June 1957

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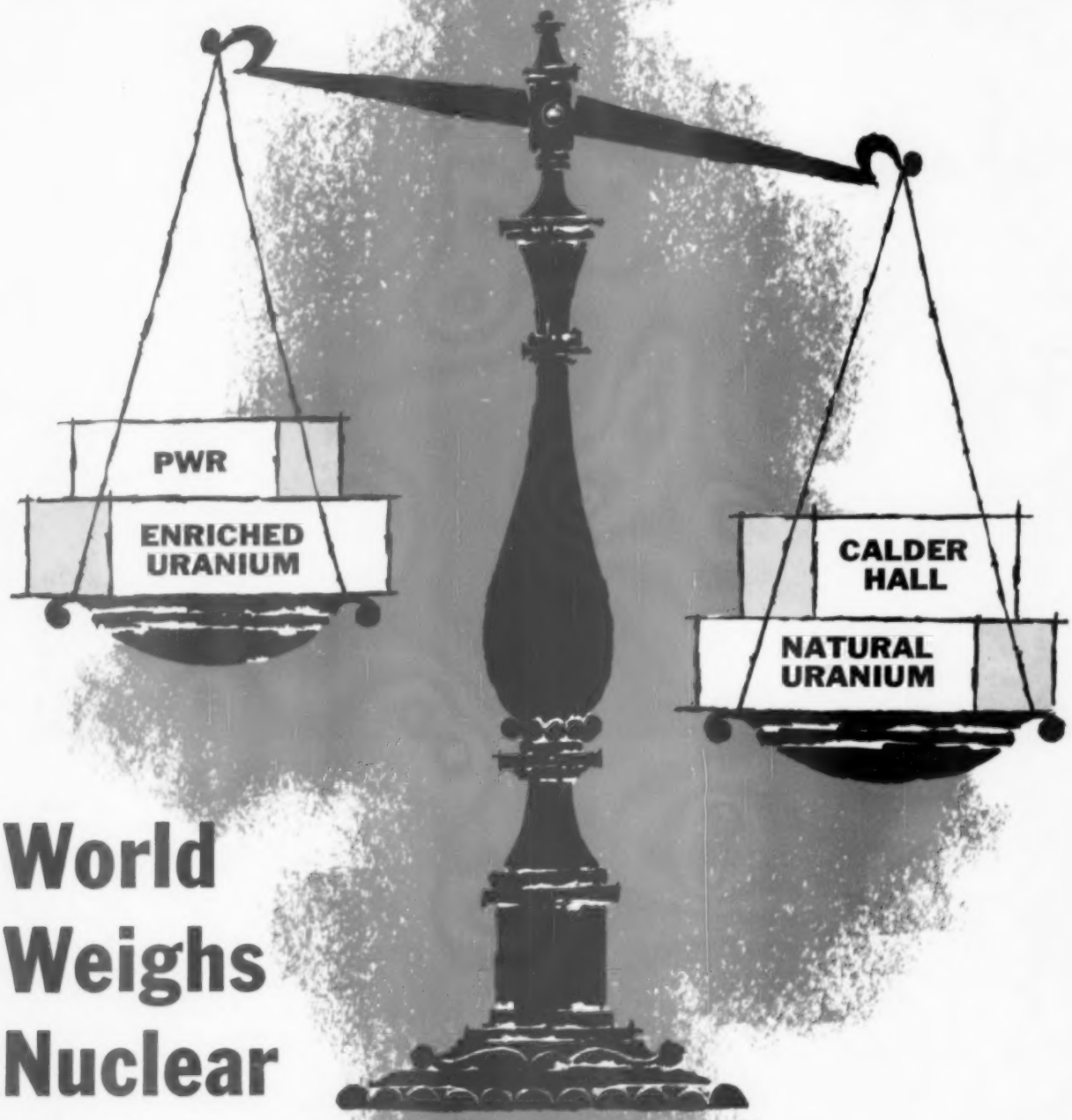
NUCLEONICS

ATOMIC POWER • NUCLEAR ENGINEERING • APPLIED RADIATION



World Weighs Nuclear Alternatives

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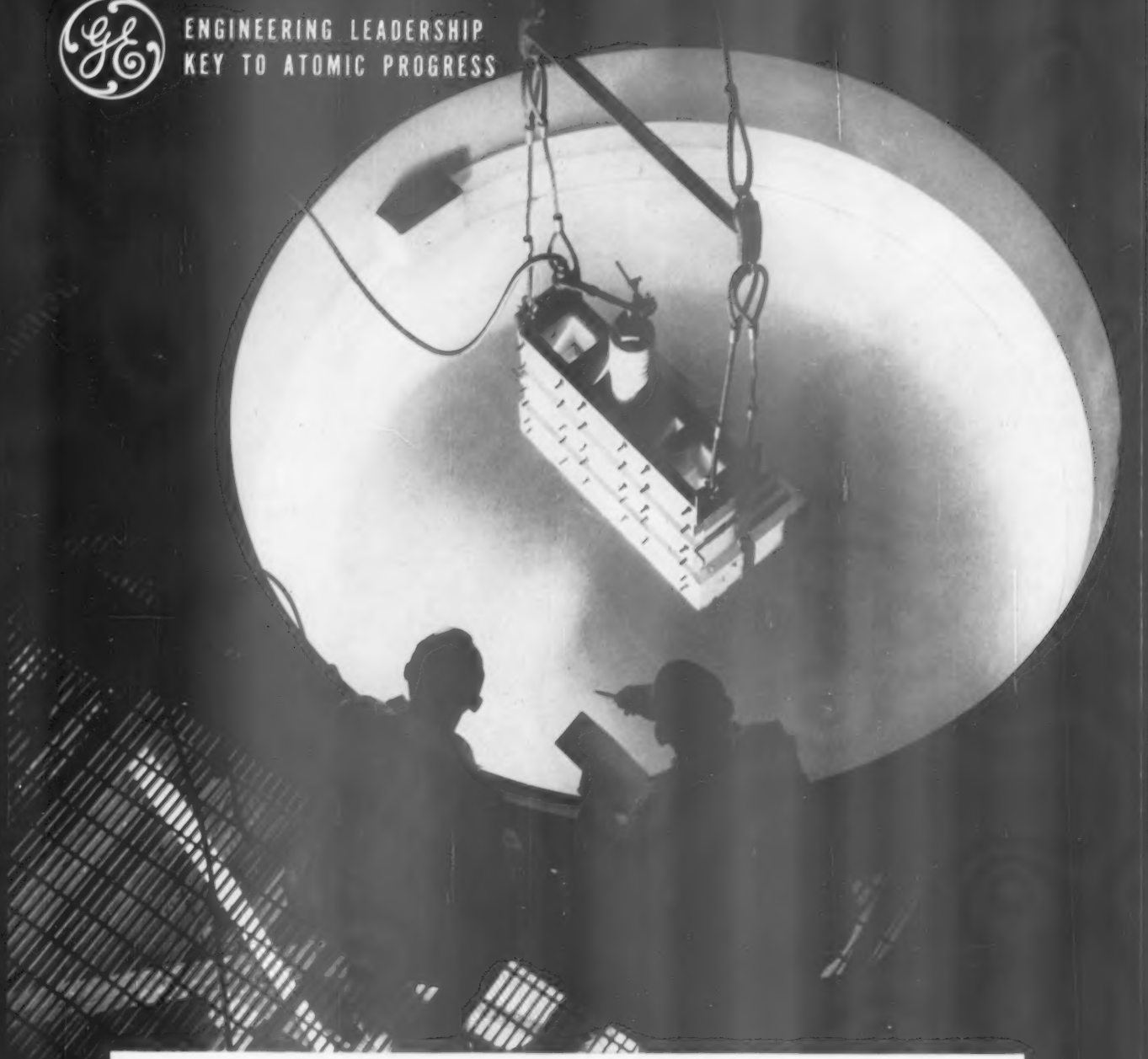
ENGINEERING LEADERSHIP
KEY TO ATOMIC PROGRESS



General Electric engineers pre-test reactor servicing techniques in 54-foot-high mock-up vessel

Reactor pressure vessel conditions, including heat and water circulation factors, are simulated in this 54-foot-high refueling test tank. With reactor core mock-ups in the tank, General Electric engineers prove refueling procedures and equipment design before the reactor leaves the drawing board.

To devise a practical method of removing the pressure vessel head closure prior to refueling, tests are now being conducted on the studs used to secure the closure. The studs are insulated and heated under water to study elongation characteristics and removal techniques.

Proving equipment design in the development stage is another example of General Electric engineering leadership in the field of atomic energy. Atomic Power Equipment Department, San Jose, California.

196-7

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NUCLEONICS

ATOMIC POWER
NUCLEAR ENGINEERING
APPLIED RADIATION

VOLUME 15 NUMBER 6

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About NUCLEONICS

Somebody up there likes us. After being misdirected to the production department because someone thought it contained advertising plates, there arrived in the editorial offices of NUCLEONICS a package containing the formal evidence of an achievement that pleases us enough to want to share it with you. Specifically the achievement is a Jesse H. Neal award for our December, 1956, report on Calder Hall and the evidence thereof is the plaque awarded each editor that worked on that report. (The Jesse H. Neal awards are made by the Associated Business Publications, an organization representing 160 publications, based upon the judgement of an independent jury.) The plaques read "for outstanding journalism" . . . which makes the team of electrical engineer, two physicists, chemist, and historian, all turned editor, that put out the Calder report, feel that they're doing well.

•

Calder continued. The Calder report, as many articles do, raised new questions even as it answered old ones. The emphasis shifted from "What is Calder?" to "What does Calder signify?" We addressed ourselves to that question with the interviews with U. S. and foreign delegates printed in our December '56 issue, but some of the answers were still being formulated . . . the delegates had not had time to evaluate Calder economics in their own countries.

Today, five months after the Calder symposium, some of the questions are finding answers. On p. 72 of this issue Dave Herron of American-Standard reports on an AEC-sponsored study of how PWR and Calder-Hall economics compare. Still on the docket are such matters as: What would be the economics of an American-designed power-only gas-cooled natural-uranium-fueled graphite-moderated reactor that could take advantage of U. S. technology and adjust to U. S. economics? Studies are being made of just that question, and when the results are available we will relay them to you.

Underlying the specific question of how Calder compares with PWR are broader questions involving the question of enriched- and natural-uranium reactor systems. Assistant Editor Hal Davis reports on a discussion of these matters held recently on page 68.

•

Meetings, meetings, meetings. About the time this issue of NUCLEONICS reaches you, June 10–12 to be exact, the American Nuclear Society will be holding its third annual meeting in Pittsburgh. A week later the Health Physics Society will hold its 1957 meeting in the same town. Few days after the Society of Nuclear Medicine will hold forth in Oklahoma City. Now, without any knowledge of the circumstances that led these societies to choose these particular dates and places, we would hazard the guess that many of their members would be better served if they were readily able to attend one of the other meetings too. That would seem to be particularly so in the case of the ANS-Health Physics near-miss, where there is a substantial overlap of interests.

We'd like to offer our good offices in this connection. Each month we carry a list of nuclear meetings under the heading of Nuclear Calendar. We would like to expand that so that it is sure to include every meeting of direct nuclear significance. Since we have got to assemble the information for Nuclear Calendar in any event, we stand prepared to serve as a clearing house for societies and organizations that want to check their proposed meeting dates against those of other groups. To make this really work will take the full cooperation of all the groups having nuclear interests, which is a lot. To get this clearing house operation going we plan to write the secretaries of all the nuclear groups we're aware of, but you can help if you are planning a meeting by letting us know of your intentions as far in advance as is possible. If all meetings can be checked against our lists at the time that they are in the "gleam in the eye" stage there would be hope for a more reasonable schedule.

We don't mean to arbitrate between parties nor influence the choice of places or times but do feel that having this type of information available will enable the final choice to be made with more information than has been available heretofore. Please write Nuclear Calendar Editor, c/o NUCLEONICS, 330 W. 42nd St., New York 36, N. Y.

—The Editors

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Coming Features in NUCLEONICS

Next Month - EBWR Design and Construction

The Nuclear Navy

UO_2 as a Nuclear Fuel

Later Months - Nuclear Rockets

Food-Irradiation Reactors

Latest on Hot Labs

designed for people who must work in a controlled atmosphere

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Type R50 (±.002% at 15° to 35°C)
Double triode and 5 pigtail parts required
Input, Tube heater voltage and B voltage
Output, approx. 5V into 200,000 ohms

FREQUENCY STANDARD
TYPE 50L

Size 3¾" x 4½" x 5½" High
Weight, 2 lbs.

Frequencies: 50, 60, 75 or 100 cycles
Accuracies:—
Type 50L (±.02% at —65° to 85°C)
Type R50L (±.002% at 15° to 35°C)
Output, 3V into 200,000 ohms
Input, 150 to 300V, B (6V at .6 amps.)

*3½" high
400 to 500 cy.
optional

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TYPE 2003

Size 1½" dia. x 4½" H. Wght. 8 oz.*

Frequencies: 200 to 4000 cycles
Accuracies:—
Type 2003 (±.02% at —65° to 85°C)
Type R2003 (±.002% at 15° to 35°C)
Type W2003 (±.005% at —65° to 85°C)
Double triode and 5 pigtail parts required
Input and output same as Type 50, above

FREQUENCY STANDARD
TYPE 2005

Size, 8" x 8" x 7¼" High
Weight, 14 lbs.

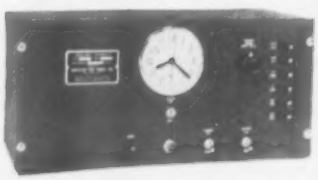
Frequencies: 50 to 400 cycles (Specify)
Accuracy: ±.001% from 20° to 30°C
Output, 10 Watts at 115 Volts
Input, 115V. (50 to 400 cycles)

FREQUENCY STANDARD
TYPE 2007T
TRANSISTORIZED

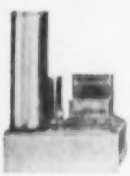
Size 1½" dia. x 4½" H. Wght. 7 ozs.*

Frequencies: 240 to 1000 cycles
Accuracies:—Same as 2003, above
Type 2007S—Silicon type
Input, 28V.
Output, Multitap, 75 to 100,000 ohms
*3½" in 2007S, 400 to 800 cycles.

FREQUENCY STANDARD
TYPE 2121A



Size
8¾" x 19" panel
Weight, 25 lbs.
Output: 115V
60 cycles, 10 Watt
Accuracy:
±.001% from 20° to 30°C
Input, 115V (50 to 400 cycles)



FREQUENCY STANDARD
TYPE 2001-2

Size 3¾" x 4½" x 6" H., Wght. 26 oz.

Frequencies: 200 to 3000 cycles
Accuracy: ±.001% at 20° to 30°C
Output: 5V. at 250,000 ohms
Input: Heater voltage, 6.3 - 12 - 28
B voltage, 100 to 300 V., at 5 to 10 ma.

FREQUENCY STANDARD
TYPE 2111C

Size, with cover
10" x 17" x 9" H.
Panel model
10" x 19" x 8¾" H.
Weight, 25 lbs.
Frequencies: 50 to 1000 cycles
Accuracy: (±.002% at 15° to 35°C)
Output: 115V, 75W. Input: 115V, 50 to 75 cycles.

ACCESSORY UNITS
for TYPE 2001-2

L—For low frequencies multi-vibrator type, 40-200 cy.
D—For low frequencies counter type, 40-200 cy.
H—For high freqs, up to 20 KC.
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P—Power supply.

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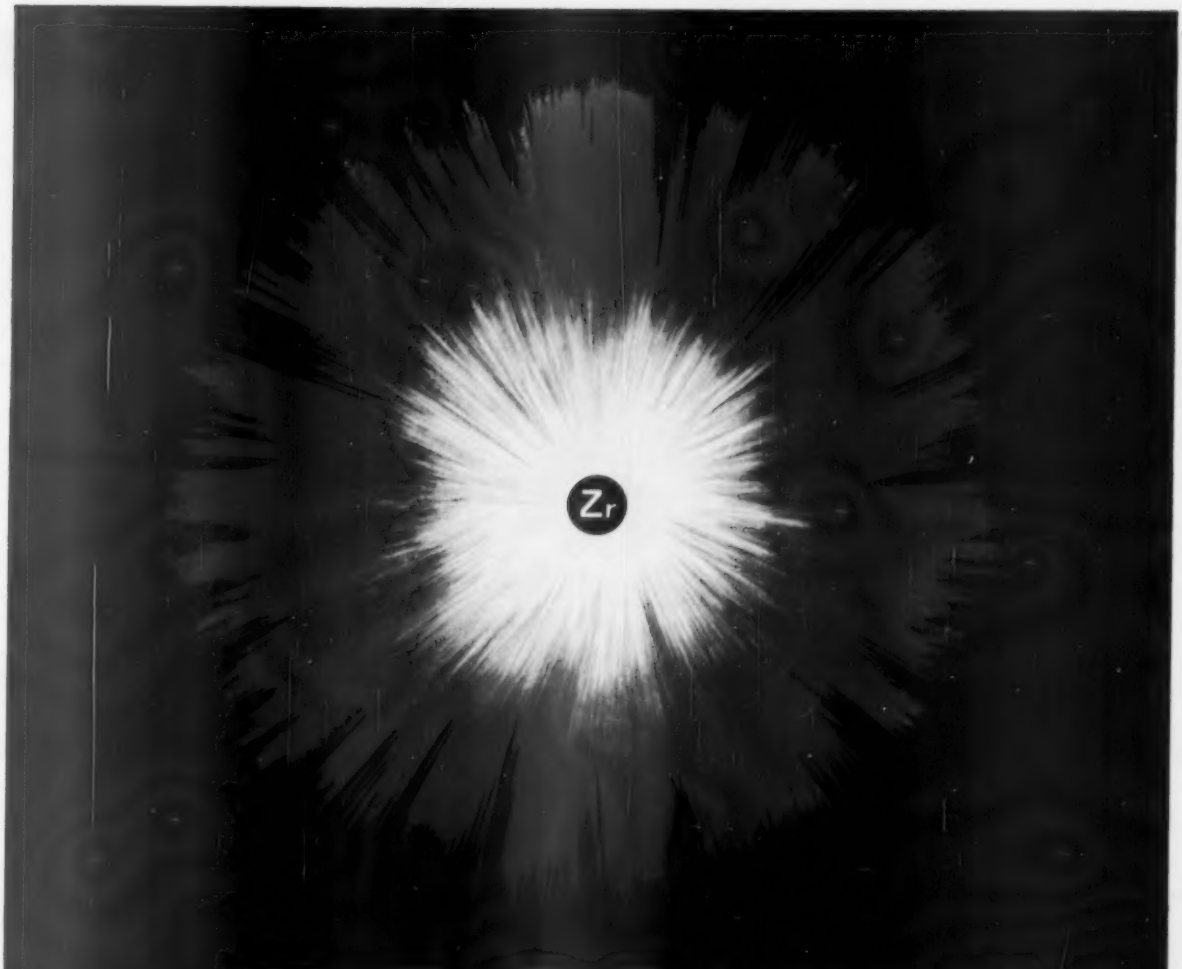

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- **Lower power generation cost**
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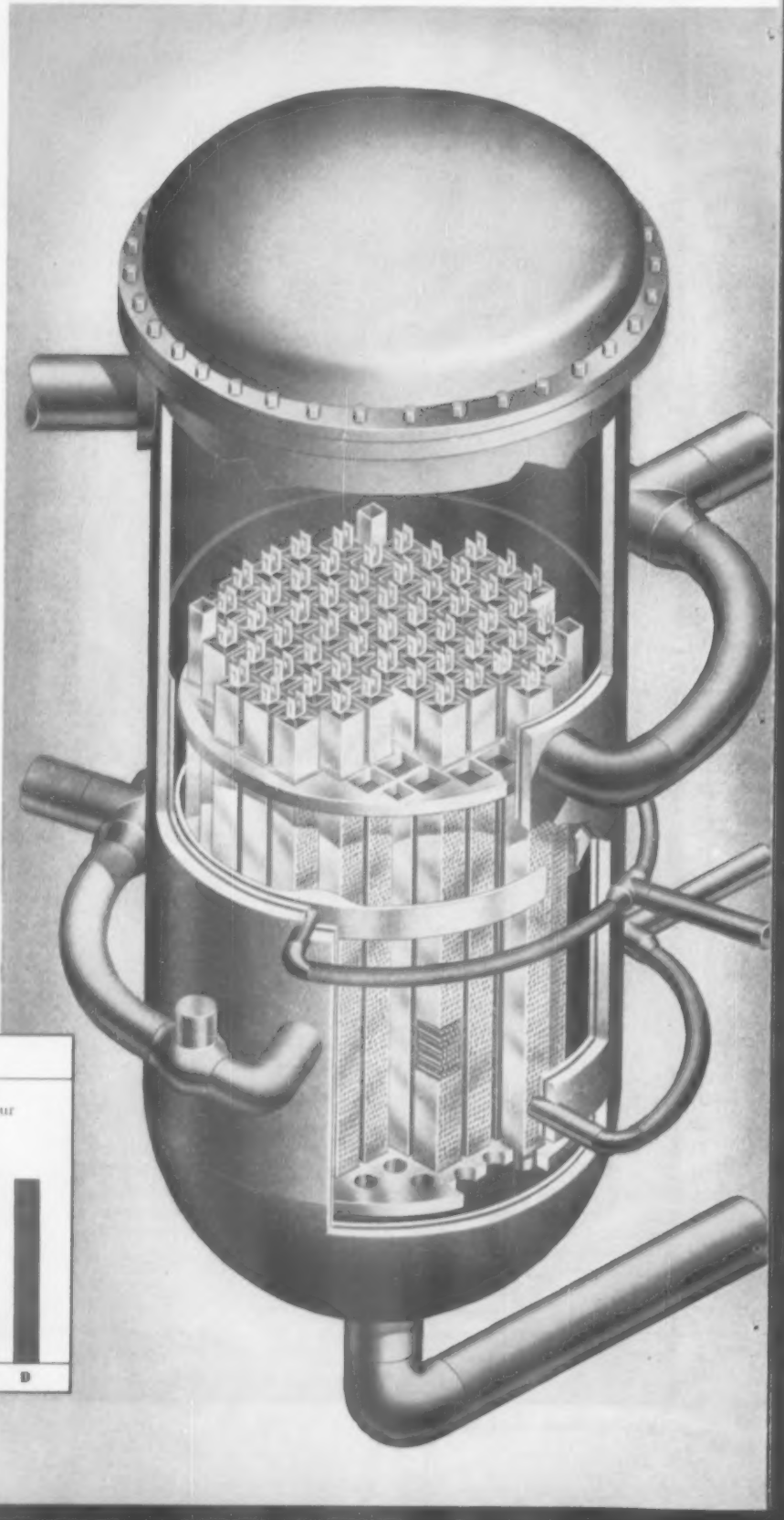
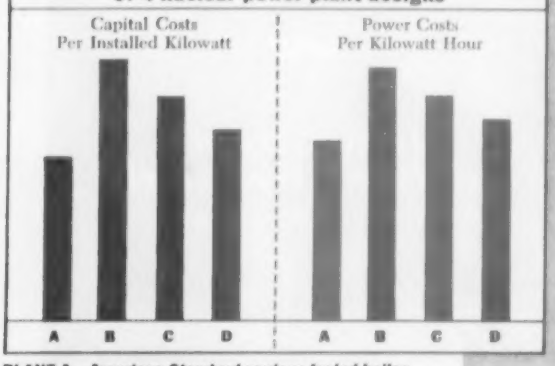
- **Simple, reliable operation**
- **Inherent safety**
- **Maximum on-line time**

** Patents applied for.*



COST COMPARISONS
of 4 nuclear power plant designs



PLANT A—American-Standard nuclear-fueled boiler
PLANT B—Pressurized-water reactor #1
PLANT C—Pressurized-water reactor #2
PLANT D—Boiling water reactor

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The American-Standard nuclear-fueled boiler employs a variable moderator level for control, with heat generation accomplished in a coolant recirculation system.

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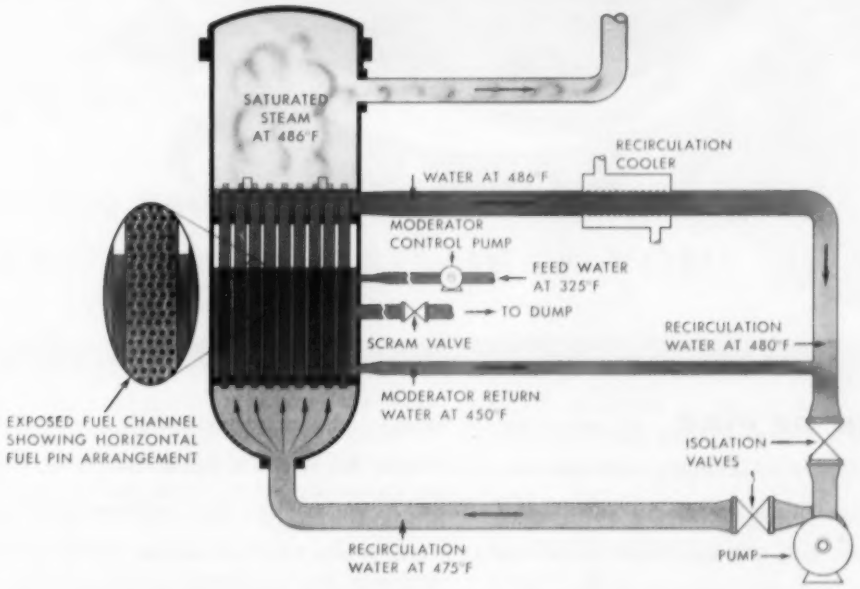
The nuclear-fueled boiler combines variable moderator control with conventional steam power plant design practices to produce a unit suited to a wide range of power requirements.



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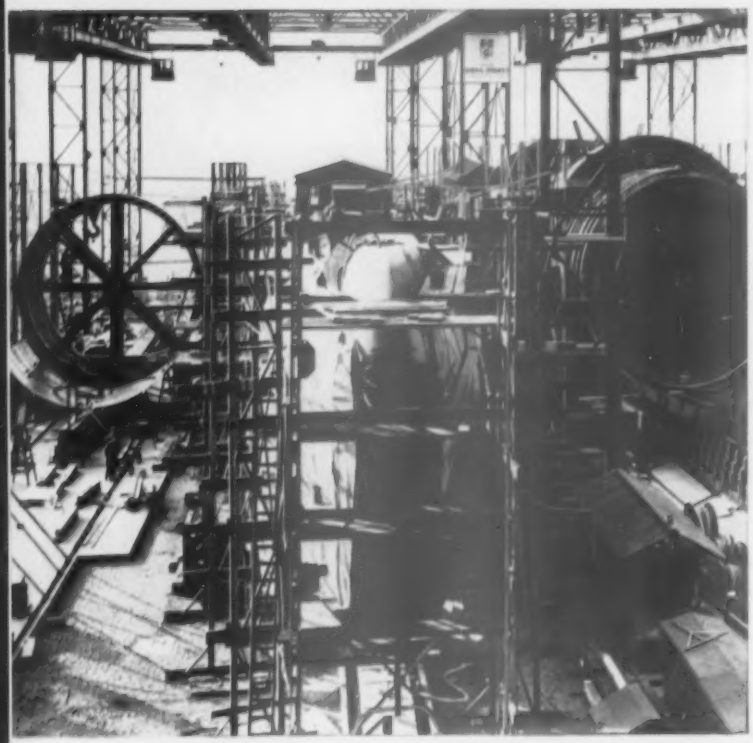

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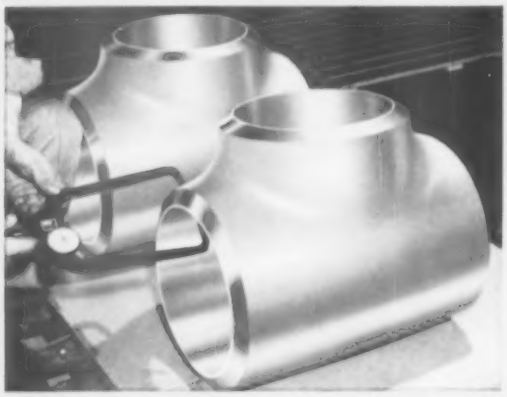
SEAMLESS STAINLESS CLAD . . . another Tube Turns' first. These heavy-wall 52° elbows for a pressurized water reactor are typical of Tube Turns' pioneering line of seamless stainless-clad carbon steel fittings.



STAINLESS STEEL. These TUBE-TURN stainless steel welding tees are for a two-reactor submarine.

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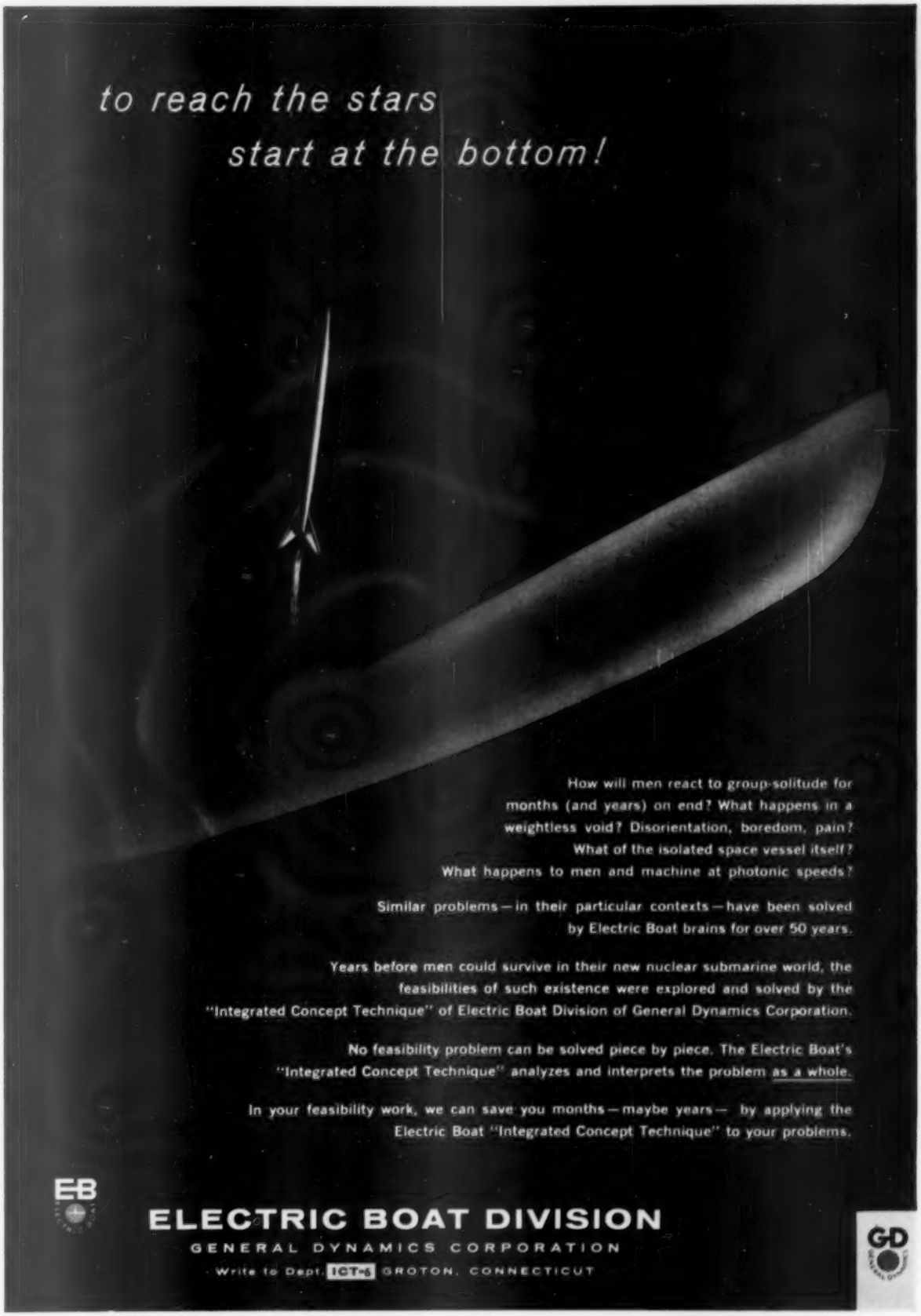
to reach the stars
start at the bottom!



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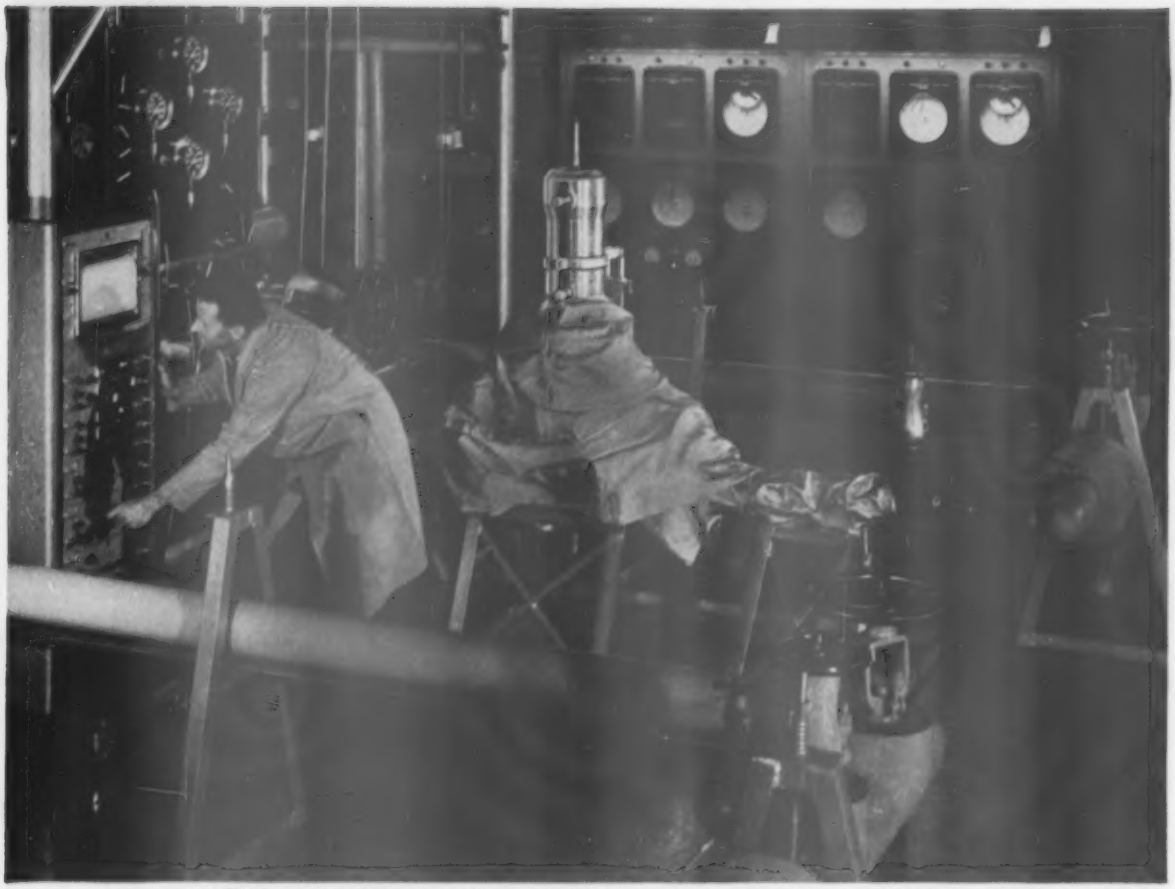

This Crane atomic valve, wrapped in insulating blanket of aluminum and spun glass, is being tested on new Crane hot test loop.

CRANE Leads in Testing Atomic Valves with New Hot Test Loop

Shown here in regular service is Crane's new hot test loop—built to control the quality and safe performance of Crane valves for atomic service.

The loop is code-designed for production and engineering testing, and operates at 550°F. It tests for leaks at seats, in stuffing boxes, in pistons and for other conditions which may occur in pressurized water reactor systems.

The loop is made entirely of stainless steel to prevent any possibility of contamination from corrosive water. Distilled water, used in all tests—hot or cold—is recirculated and repurified continuously. Flow is maintained by an hermetically sealed, canned rotor pump which has no stuffing box. Precision instruments maintain constant water level, temperature and pressure.

The new hot test loop is another Crane-pioneered testing facility that makes Crane's experience—and products—of more value if you work with atomic power, or if you operate in other fields involving control over the flow of fluids in radioactive processes.

Write to the address below for a copy of this Crane folder on valves for atomic services.

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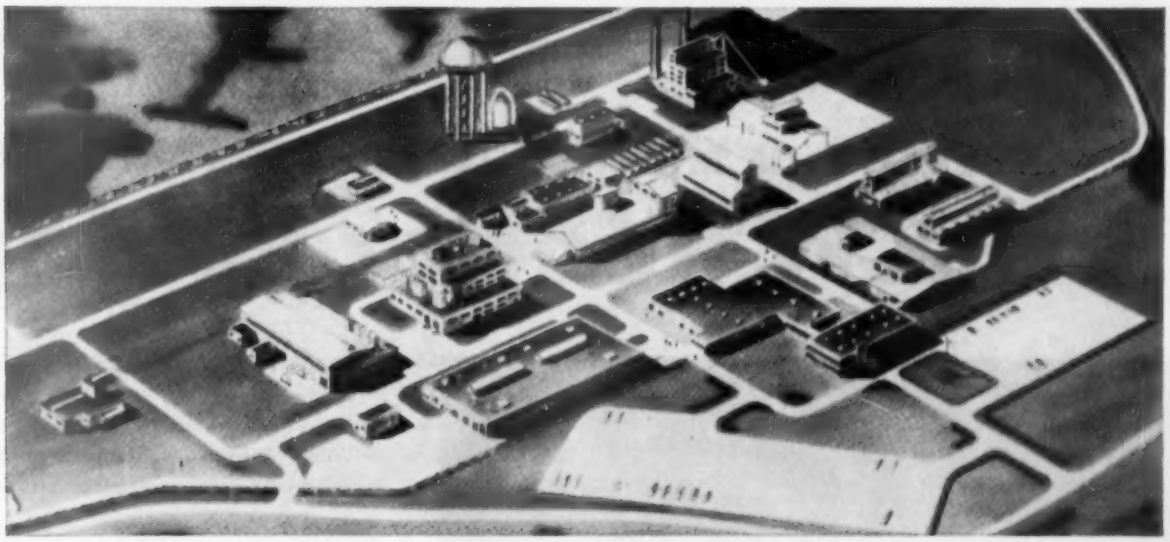
NEW

reactor feed-material plant for AEC—designed and engineered by Blaw-Knox



Lower costs in processing uranium ore and preparing the metal for use in reactors are expected from the operation of this new $50 million reactor feed material plant. Designed and engineered by Blaw-Knox, using a process developed by the Mallinckrodt Chemical Works, the plant will be operated by Mallinckrodt for the Atomic Energy Commission at Weldon Spring, Missouri.

This project is another good example of Blaw-Knox experience on a wide variety of nuclear energy projects. And it provides further evidence of why we can effectively serve you as engineers and constructors in—scoping the project • preparing feasibility reports • summarizing hazards reports • detailing engineering • procurement and fabrication of parts • construction of the facility • initial operation • training of operators.

We invite your inquiry on how we can put this experience to work on your specific requirements in this active field of nucleonics.

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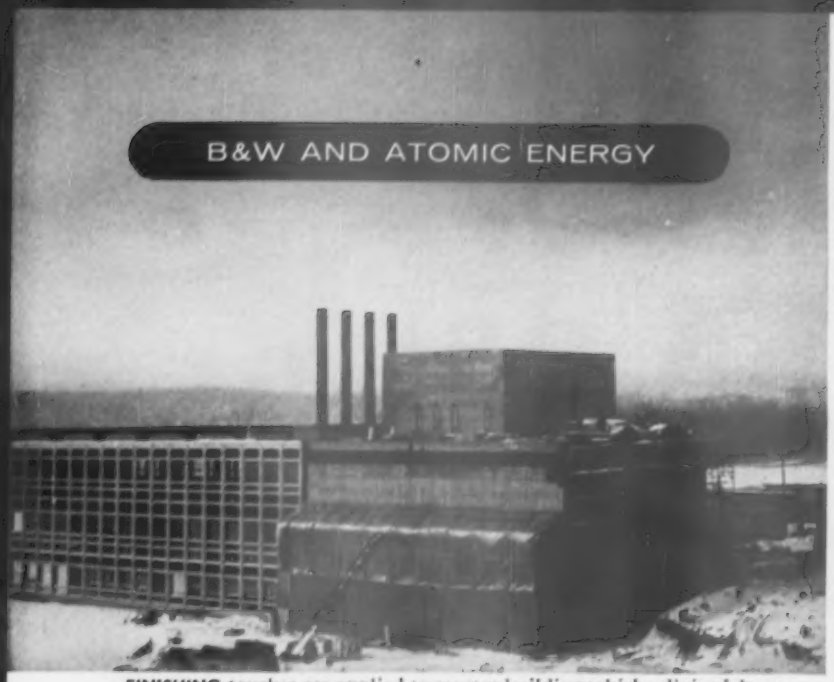
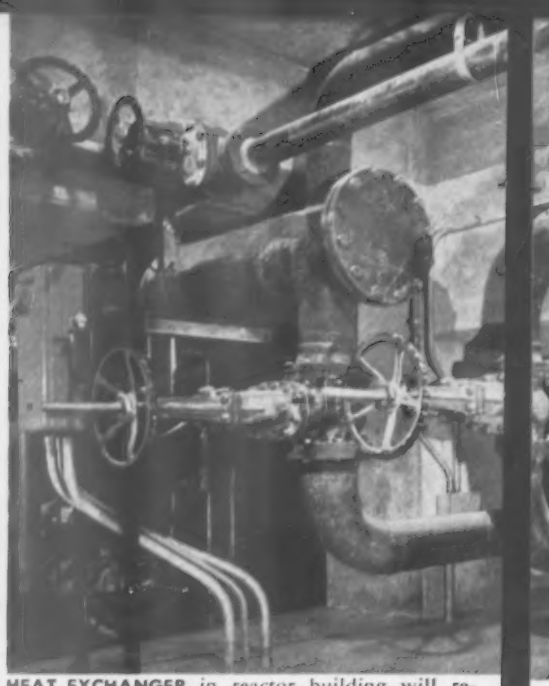


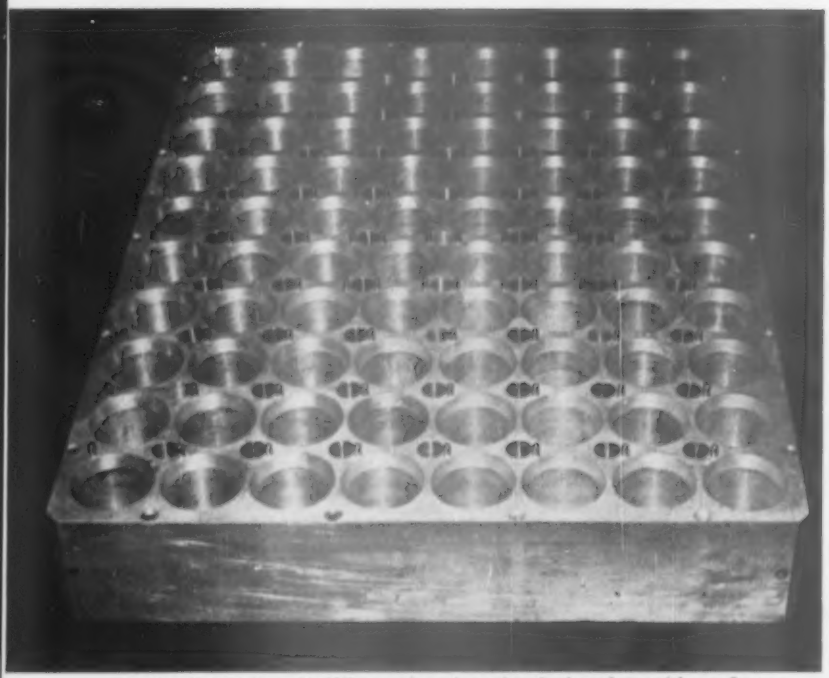
FINISHING touches are applied to reactor building which adjoins laboratory. Doors and building dampers are fitted with gaskets to provide a tight seal when closed.



HEAT EXCHANGER in reactor building will remove one megawatt of heat by forced circulation.

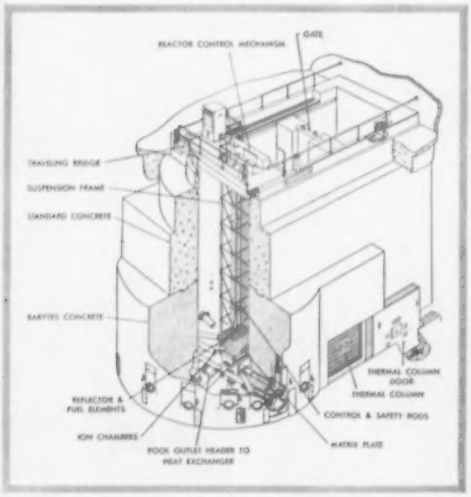
UNIVERSITY OF MICHIGAN LABORATORY ATTACKS

B&W Reactor Putting

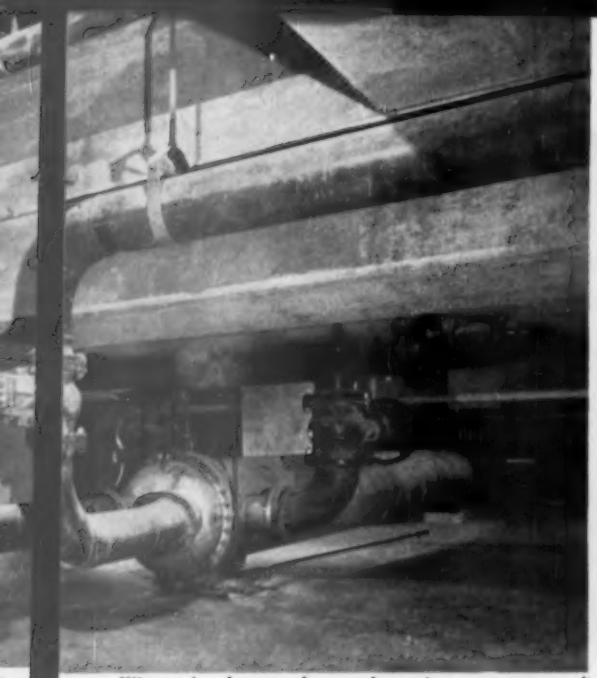


GRID PLATE has 80 different locations for fuel and graphite reflector elements, permitting many different core arrangements. Grid plugs for unused holes prevent the coolant from by-passing the fuel elements.

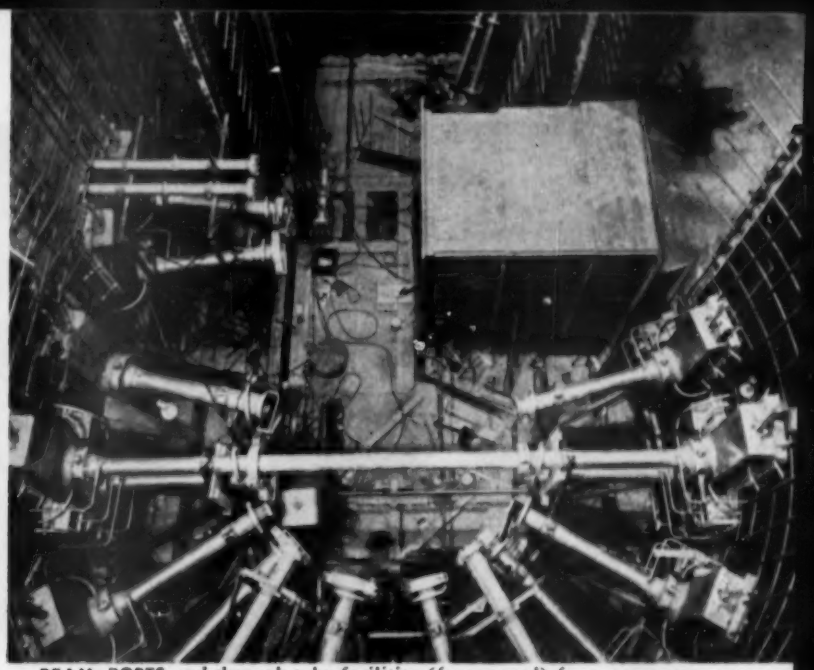
The Ford Nuclear Reactor provides an intense source of neutron and gamma radiation for basic medical, biological, chemical, physical, and engineering research and prac-



DETAILS of construction of the B&W Swimming Pool Type Reactor at the University of Michigan are shown in the cutaway drawing.



Water is drawn down through core, pumped through heat exchanger, and then returned to pool.

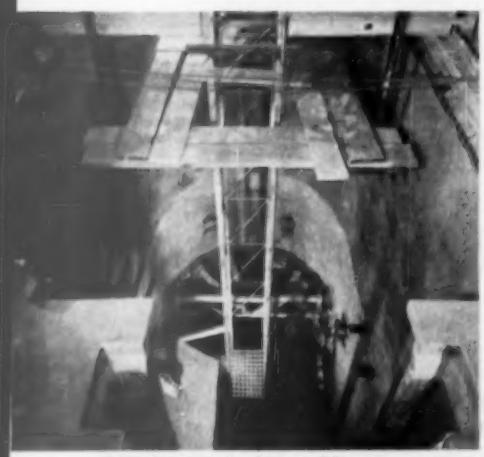


BEAM PORTS and through-tube facilities (foreground) focus on reactor core full-power operating position. Thermal column (upper right) is stacked with graphite blocks to slow down entering neutrons to thermal velocities.

BASIC RESEARCH PROBLEM WITH . . .

Neutrons to Work

tical training in nuclear technology for both engineers and scientists. It will also supply some of the radioactive isotopes for the many studies now employing tracer techniques.



GRID PLATE is in position to support fuel and reflector elements in rounded end of pool. Concrete channels in foreground are for pool gate which is raised by crane to permit passage of suspension frame.

In this design the basic pool-type reactor concept was adapted by B&W engineers to the special needs of the University of Michigan. The equipment components were fabricated and installed by B&W.

FROM THE BEGINNING of atomic energy, at Chicago, Oak Ridge and Hanford, to the application of nuclear energy to steam generation, like the new Consolidated Edison plant soon to rise at Indian Point, B&W has had an important and responsible role in the field of atomic energy. It continues to maintain its leadership by backing its long experience in related fields with a continuing, independent research and development program that studies the implications and the uses of a whole new world of energy.

FOR NEARLY 100 YEARS, B&W has been intimately identified with power generation, corrosion research, heat transfer problems, high temperature metallurgy, steam technology, and fabrication of heavy pressure vessels. The introduction of atomic energy found B&W engineers particularly equipped to cope with the problems posed in this new field. It was, in effect, an extension of familiar problems which they were called upon to answer.

Whether it is a research reactor to fit a research budget or a complete nuclear steam generation plant, with a single responsibility from fuel element to turbine throttle, consult The Babcock & Wilcox Company, Atomic Energy Division, 161 East 42nd Street, New York 17, New York.

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ATOMIC ENERGY DIVISION

EBWR control room features Speedomax instruments for critical plant parameters.

L&N INSTRUMENT SYSTEMS

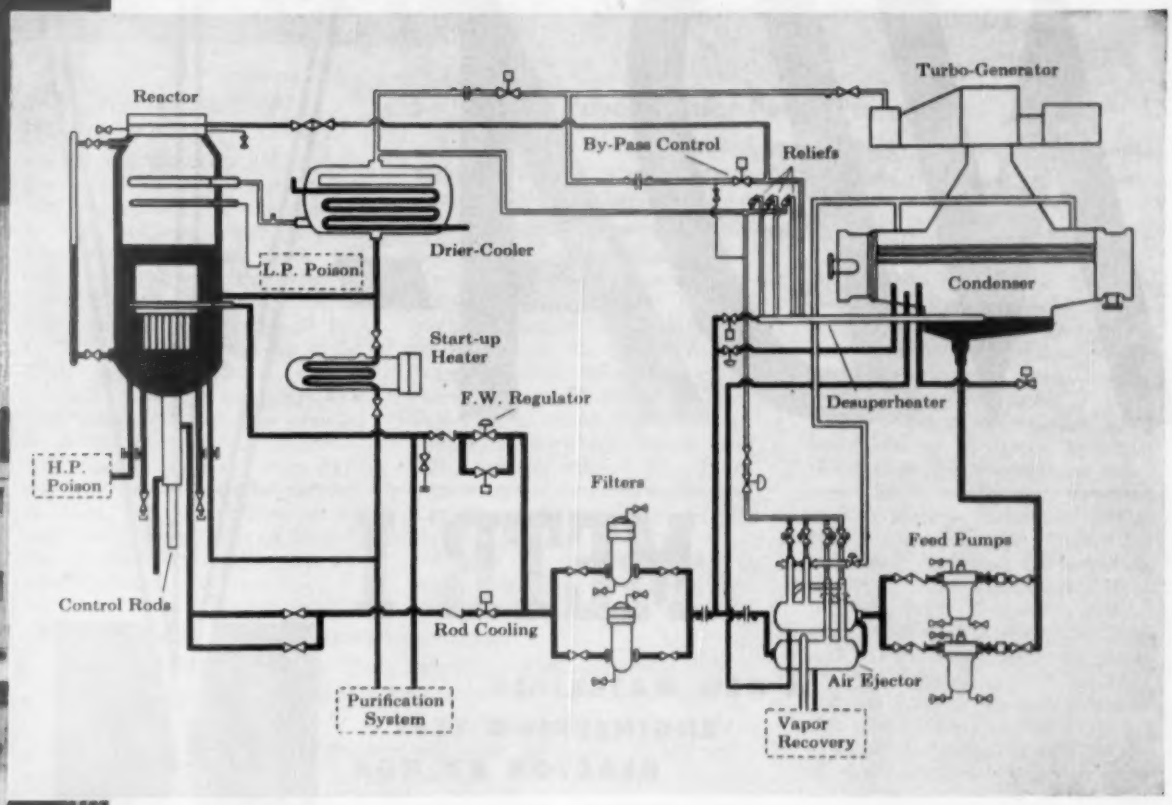
EBWR makes

Ever since initial start-up of the Experimental Boiling Water Reactor, early this year, the generation of 5000 kw for the electric system of Argonne National Laboratory has been directed through the reactor's L&N instrument systems.

These instruments represent the basic L&N approach to nuclear power instrumentation—standardized components, engineered into complete systems by men with electric power *and* nuclear control experience.

In EBWR this combination resulted in a

Photos Courtesy Argonne National Laboratory



EBWR flow diagram *illustrates the basic simplicity of this reactor type. Water is converted to steam within the reactor vessel itself. This eliminates heat exchange equipment. The steam flows to the turbo-generator where electricity is produced, then through the condenser, filters, etc., back into the reactor vessel.*

GUIDE NUCLEAR POWER PLANT OPERATORS AS

Megawatts from Neutrons

coordinated, all-electric, plant instrumentation arrangement that includes a complete three-element feedwater control system.

Data on neutron flux is fed to Speedomax® recorders mounted at the EBWR operating console. The L&N feedwater system controls the flow of water into the reactor, either to hold level at a fixed point or to allow level to deviate sufficiently to hold water flow equal to steam flow. Other temperatures, pressures, flows, and radiation levels are monitored by Speedomax instruments around the control room.

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METR I

A NEW MATERIALS ENGINEERING TEST REACTOR BY *NDA*

NDA has now completed preliminary design of Materials Engineering Test Reactor I for construction at Mol as Belgian Reactor II. Joined by senior personnel from CEAN, *NDA* is now carrying out detailed design. *NDA* believes METR I will be of interest to nuclear engineers and scientists throughout the world. Consider these characteristics:

- **Thermal power level: 50 MW**
- **Available core neutron flux during experiments**
  Fast . . . . . 2×10^{15} n/cm²/sec
  Thermal . . . 6×10^{14} n/cm²/sec
- **Single fluid—light water cooling**

Consider also the flexibility of METR I:

- **Five 8½" through holes in core section**
- **Central irradiation tubes in each fuel element**
- **Forty 3½" holes in core matrix (including 10 through holes)**
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- **Seven radial beam holes up to 12" in diameter**
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With its high available neutron flux, flexibility and ease of access, METR I is unsurpassed as a test reactor.

Opportunities exist at *NDA* for qualified scientists and engineers interested in reactors of advanced design for research, power and propulsion.

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ROUNDUP of Key Developments in Atomic Energy

AEC Releases Pu Price, Hikes It by 150%

AEC will pay from $30 to $45/gm for plutonium produced in private reactors between now and July 1, 1962, the Commission announced. The newly-declassified price will vary within this range depending on the Pu^{240} content of the material. For the subsequent year until July 1, 1963, the price will be a flat $30/gm. AEC said it expects gradually to reduce the price "to a level based upon the fuel value of Pu in commercial power reactors."

The earlier price had been $12/gm, but this was rescinded last fall. At least one prospective reactor operator, Walker Cisler of Detroit, had complained to Congress that the Pu price was too low.

$10-Million Private Fusion Power Project Set

A landmark in development of atomic energy was passed last month with the entry of private industry into the field of controlled thermonuclear reseach, with private money.

A $10-million, four-year basic research program will be carried out at the new Hopkins Laboratory for Pure and Applied Science of General Dynamics Corp.'s General Atomic division at San Diego, Calif., on behalf of the Texas Atomic Energy Research Foundation. The Texas group of 11 utilities did not wait long to make big news: its announcement—which caught the utility industry entirely by surprise—came only one month, to the day, from the group's establishment (NU, May '57, 28).

At that time spokesmen said "the program of the Foundation is long-range, since the natural gas now used in the Southwest will be competitive with nuclear energy for many years." But few, then guessed how very long-range indeed was the program being considered.

Aim of the research project is "an understanding of heavy hydrogen reactions at high temperatures under controlled conditions." Research work is already under way at San Diego, where General Atomic has assembled more than 150 scientists, engineers and technicians. It is understood the $10-million cost of the project will be shared by General Dynamics and the Texas group. No government funds are involved.

Frank Pace Jr., new president of GD, told NUCLEONICS that it was not expected the project would get out of the laboratory stage into pilot models during the life of the contract, although that is of course the eventual goal.

Although the main Hopkins Laboratory building—planned with lots of room for different new projects—will be where most of the work will take place, GA does intend to erect a small specialized structure: an auxiliary building to house special experiments in connection with the work. Here will be placed any apparatus it is decided to mock up.

W. A. Parish, president of both Houston Lighting and Power Co., and of the Research Foundation, said the project would strive first to demonstrate controlled fusion conditions "without immediate prospects for economical energy production from this source." He pointed out that if, ultimately, the latter could be achieved, it promises not only an almost unlimited source of energy in deuterium from seawater, but also emancipation from the problems of large-scale radioactive waste disposal. He said the decision to proceed was based on the belief the Texas region would "probably be the last section of the nation that would ever use the fission process, so we decided to explore all possible avenues."

Although AEC chairman Strauss

Martin to build Brazil a two-reactor station

Power-hungry São Paulo State, Brazil's most industrialized state, has decided to build a 25-Mw nuclear power plant at Jurumirim Sorocabana, about 50 mi W of São Paulo city. The Martin Co. will supply two 12.5-Mw pressurized water reactors. So learned NUCLEONICS' Rio bureau in an exclusive interview with Mario Lopes Leão, head of São Paulo state power companies, Cia. Usinas Eletricas de Paranapanema and Cia. Hidro-Eletricas do Rio Pardo. These firms will be shareholders in a new private company being formed to run the plant. The project will cost $18.75-million—$750/kw installed. Preliminary engineering is to begin next month. U. S.-Brazilian negotiations for a power bilateral are nearing completion.

Asked why build two 12.5-Mw reactors rather than one 25-Mw unit, Martin nuclear chief Tibor Nagy told NUCLEONICS the answer is rain. Brazil, largely dependent on hydro power, is subject to cyclic rainfall. In dry season, when reservoir levels drop, both reactors would run; in transition periods, one could meet the load; maintenance and refueling shutdowns would be scheduled when reservoirs are full. The project would give Brazil two private nuclear power plants (the other being American & Foreign Power: NU, April '57, 19), and one research reactor in operation by year's end, putting her far out in front in Latin America in atomic energy. The 5-Mw pool research reactor, built by Babcock & Wilcox for the University of São Paulo, will go critical in September—first reactor in South America.

AMF Atomics sells Israel, Univ. of Florida

Israel is buying a 1-Mw pool research reactor from AMF Atomics, according to Yehudah Yeivin of the Weizmann Institute. AMF Atomics has also sold its first Argonaut-type training reactor to the University of Florida. A modified Argonaut designed by Walter H. Zinn's General Nuclear Engineering Corp., it has a square instead of circular core, and will be able to operate continuously at 10 kw instead of only intermittently.

had told a press conference on April 1 that "there has been nothing yet that could be described as a major break-through," the Texas announcement set off a new rash of optimistic savoring of the promise in the arithmetic of fusion (total fusion of 1 kg deuterium would produce 100,000,000 kwh of electricity; fusion of all the deuterium in the oceans would provide man with energy for a billion years).

The work at San Diego will be subject to AEC classification, as on Project Sherwood, in keeping with the provisions of General Atomic's access permit as well as the broader requirements of the Atomic Energy Act. However, there is a real possibility that patentable ideas would emerge from the research, which would be of considerable significance particularly if fusion power were to become a reality earlier than is generally anticipated.

General Atomic has unique qualifications for the work being undertaken. Frederic deHoffmann, manager, played a key part in thermonuclear calculations ancillary to the conception and construction of the H-bomb. Research Director Edward Creutz, formerly head of the physics department at Carnegie Tech, spent a year (Aug. '55–July '56) visiting and working at the major Project Sherwood sites as part of a study of its activities conducted for the AEC. Marshall Rosenbluth, formerly with Los Alamos, is credited with major contributions to the theory of the pinch effect.

Other General Atomic personnel assigned principally to the new thermonuclear effort are Charles Oxley, Samuel Cunningham (formerly with Project Matterhorn), David Brower, David Rose, Harold Thomas and Norman Rostoker. However, GA officials point out that in this, as in their other projects, all members of the laboratory will be called upon for assistance. They see the project rapidly coming to absorb about a third of their total effort.

In addition, consultants will be brought in to advise on this and other problems; prominent names here include theorists Kenneth Case, Freeman Dyson, and Felix Adler.

The private project is expected to have little influence on the direction and pace of AEC's own controlled fusion program, Project Sherwood (NU, Nov. '55, 62; Feb. '56, 42; June, 36; May '57, 108).

Two 3d-Round Bids for AEC, New Job for Westinghouse

Two southern utility combines plodding the road to nuclear power last month passed milestones; as a result, AEC got its first formal third-round proposal, Westinghouse chalked up another major project, and Walter H. Zinn's General Nuclear Engineering Corp., retained by both groups, found itself in the reactor consultant business with more than two feet.

The same week, a second proposal for AEC third-round aid came from Northern States Power and ten associates in Central Utilities Atomic Power Associates, who are planning a boiling reactor now up from 60- to 66-Mw.

• First proposal under the third round of AEC's Power Demonstration Reactor Program came from Florida Nuclear Power Group (NU, April '56, 22). The three Florida utilities are seeking a Savannah River type reactor harnessed for power: natural uranium fueled, heavy water moderated, with gas cooling. This is one of two large reactor types to which AEC promised priority under the third round's ground rules (NU, Jan. '57, R1). The proposal contemplates a 136-Mw net (electric) unit, to cost $40-million; AEC is asked for research and development assistance and waiver of fuel use charges for the reactor's first five years. The group has engaged GNEC to design the reactor and to carry out research and development work including development of the natural uranium fuel elements. A year ago, the then-new Florida combine said it had selected Babcock & Wilcox to design and build the reactor; an LMFR was then mentioned as a possibility.

• Carolinas-Virginia Nuclear Power Associates (NU, Nov. '56, R6; Feb. '57, R1) entered into an agreement with Westinghouse to develop its nuclear plant. The four tidewater utilities had earlier chosen Stone & Webster as engineering consultants and GNEC as reactor consultants. Reactor type was not specified; target date for completion is 1962. CVNPA says it has not yet decided whether or not to make a proposal for third-round aid. For Westinghouse, it is the fourth major domestic nuclear central station project (PWR, Yankee, and Pennsylvania Power & Light's big aqueous homogeneous are the others), plus power units for Belgium and Italy, and naval propulsion plants.

Other Projects' Progress

An informal NUCLEONICS poll of other groups that had announced reactor plans turned up little progress—or at least little that the planners were willing to talk about.

• Middle South (NU, Jan. '57, R1) is still in discussion stage, has "no final plans."

• New England Electric (NU, Dec. '56, R3) is understood to be waiting for the Yankee project, in which it is a principal participant, to progress further; and for selection of a site to be made.

• American Gas & Electric (NU, April '57, 24) has not finalized plans, is studying comparative fuel costs in its low-cost fossil power area.

• Pacific Gas & Electric (NU, April '57, 24) said it has nothing new to report.

• Isotope Products Inc. (NU, April '57, 17) is working out final details on financing its cobalt-60-and-space-heat reactor; feels it unlikely that AEC financial aid will be sought. GNEC is also designing this one.

• General Public Utilities' plans for a 60-Mw reactor in the Manila area in the Philippines are still in study stage. Westinghouse and Babcock & Wilcox made proposals last August to build a pressurized water plant. GPU then said it would complete its analysis of the bids by December, today says it has not yet done so.

Britain May Meet Goal With Only 10 Stations

Improvement in design of Britain's Calder-type reactors has been so rapid that the government now expects to meet its 1965 target of 6,000 Mw installed nuclear capacity with as few as 10 two-reactor stations, instead of the 19 originally planned (*see p. 27*). Contracts for further stations are to be awarded semi-annually starting in early 1958.

Japan Completes First Reactor, Gets First Enriched-U Export

Work on Japan's first research reactor, the 50-kw water-boiler supplied by Atomics International, was completed May 22. The first load of uranyl sulfate solution was expected shortly from the U. S.; it had been shipped from Mallinckrodt in St. Louis, first commercial export of enriched uranium. The reactor is scheduled to go into operation in July.

Unions Get Support At PRDC Hearing

The three labor unions contesting the construction permit issued by AEC to Power Reactor Development Co. of Detroit for the 100-Mw Enrico Fermi fast breeder last month presented their case. They got support for their position—that the reactor is unsafe at that location on the basis of present knowledge—from C. Rogers McCullough, AEC's top reactor hazards evaluation officer and chairman of its Advisory Committee on Reactor Safeguards, and from D. A. Rogers of Allied Chemical & Dye Corp., a member of ACRS. Both men held fast to the position ACRS took last June in a letter from McCullough to AEC stating that "there is insufficient information available at this time to give assurance that the PRDC reactor can be operated at this site without public hazard."

Also testifying were AEC general manager Kenneth Fields, and Harold L. Price, director of civilian application for the Commission. Union counsel Benjamin Sigal tried to show in his questioning of Fields and Price that beyond the published regulations, AEC had no standards, codes or criteria by which it judged license applications; that this means in practice that AEC need follow no rules at all, and that licenses can be granted arbitrarily.

AEC, third party to the hearing, will present its case early in June.

Compromise Would End Acceleration Deadlock

An attempt at a compromise solution of the civilian reactor acceleration debate in Congress was offered by Rep. James E. Van Zandt (R-Pa.), member of the Joint Committee on Atomic Energy.

His bill (HR 7472) would: (1) authorize appropriation of $100-million for construction, at AEC sites, of advanced design and development reactors up to 50 Mw, electricity generated to be used by AEC; (2) provide additional statutory authorization for AEC to give more aid to private industry in the construction of prototype power reactors; (3) authorize AEC to construct supporting facilities for the program including fuel fabrication and reprocessing plants; (4) require that information on full amount of AEC financial aid be disclosed—when disclosure does not endanger military security—in order that Congress and the public be able to determine relative costs and economic benefits; (5) provide that the program must not interfere with weapons production; (6) provide that reactors built under this program be sold to the highest bidder after 15 years' operation; (7) offer rapid tax write-off privileges to builders of such reactors.

Van Zandt said his bill did not authorize a crash program for government construction of large-scale power reactors because that "is the proper province for private industry." His program "should not hurt the coal industry," he said. (Last year 18 coal-state Democrats voting against the Gore bill provided the margin of defeat, and Van Zandt is from a coal state.) The bill's author sees it as a "sound" compromise, acceptable to public and private power proponents, between the Democrats' $400-million crash program of large reactor building by AEC, and those who would leave everything to private industry. However Rep. Chet Holifield (D-Calif), House sponsor of the Gore bill, called the Van Zandt proposal "too little and too late."

Pebble-bed reactor: high-quality steam, easy fuel

Another promising new reactor system—this one offering steam conditions and thermal efficiencies equal or superior to the most modern conventional plants, plus negligible fuel fabrication costs—is being developed by Sanderson & Porter, New York engineering firm, in collaboration with Alco Products. This is the pebble-bed reactor: a tube or column filled with spherical pellets of enriched uranium carbide and graphite homogeneously mixed. These balls, which can be used from ¼-in. to 2-in. dia, can be made with a standard pelletizing machine, to standard engineering tolerances. Based on current prices for making pure graphite balls, it would appear that the 35-ft³ column (20 kg critical mass) in a 50-Mw (heat) reactor could be filled for less than $10,000 fabrication cost—conceivably as little as $1,000. Systems are being studied for reactors ranging in size from 16.5 to 250 Mw (electric). The column is cooled by helium circulated by a standard, off-the-shelf blower—a centrifugal compressor as used in gas-line service. The helium enters at 300–400° F, leaves at 1,400–1,500° F, and produces steam ranging from 850 psi, 900° F for a 16.5-Mw reactor, to 3500 psi, 1100° F/1050° F reheat, for use in a supercritical turbogenerator unit. Instead of trying to contain the fission products, they may escape through the porous balls, to be carried off by the coolant stream and subsequently removed in reactive traps before the helium reaches the heat exchangers. High burnup—a one-year fuel life—is planned. The reactor is "loaded with a shovel at the top, and unloaded into a wheelbarrow at the bottom." The Germans are also working on this system (*see p. 25*).

IAEA first conference put off to October

The first general conference of the International Atomic Energy Agency, originally set for Aug. 19, has been postponed until Oct. 1 by IAEA's preparatory commission. Slowness of signatory nations' legislatures to ratify the Agency's statute is believed to be the reason. The U. S. Senate is still debating ratification. The conference, in Vienna, will setup IAEA and fix the first year's program.

General Atomic to test new safe concept

The new "entirely safe" reactor concept announced last fall by General Atomic (NU, Nov. '56, R8) is embodied in a critical assembly that is nearly completed, with experiments to start in a few weeks. The concept is of a solid-fuel homogeneous reactor, with a core of a solid mixture of partly-enriched uranium and some light, moderating material such as beryllium or zirconium hydride. Two buildings—one housing the 1-watt assembly, the other the control panel and workshops—are completed at GA's San Diego labs; the assembly's steel frame is up; instrumentation is in.

Oak Ridge Ran Fused-Salt Reactor Experiment in 1954, Weinberg Reveals; Promise Seen for Low-Cost Power

Until last month, the uncleared half of the U. S. atomic industry thought of fused-salt reactor systems as no more than a promising idea for the long-range future—further over the horizon than LMFR. The only references, vague and veiled, had been to "investigations" of the idea being pursued at Oak Ridge (NU, Oct. '56, R5).

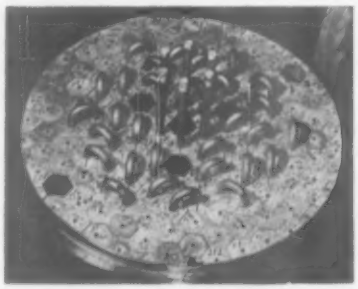
Last month Oak Ridge National Laboratory Director Alvin M. Weinberg dropped a bombshell before an unsuspecting audience of Latin American scientists invited to an international symposium at Brookhaven: the U. S. had, in 1954, built and operated a fused-salt reactor, at 2.5 Mw and 1,500° F. It had been part of the aircraft nuclear propulsion program; the secret had been well kept.

The Aircraft Reactor Experiment, as it was called, was a molten-fluoride U^{235} burner—a circulating-fuel, extremely-high-temperature reactor in which the fuel was a mixture of molten fluoride salts—NaF and BeF_2—carrying UF_4. A separate beryllium oxide moderator is required because of the low moderating power of the fuel constituents. The fuel flowed in serpentine nickel-alloy tubes through the core where the system became critical (*see flow-chart*); a high-temperature centrifugal pump circulated the liquid to a heat exchanger. A secondary circuit contained high-velocity helium—an inert material was needed because of the presence of sodium, as the second heat exchanger dumped the reactor heat to water coolant. Heat generated in the BeO reflector was removed by circulated sodium, also dumped to water by means of an intermediate helium loop.

Because of the thermal expansion of the fuel, the reactor was expected to be entirely self-controlling; the only controls, other than the inherent negative temperature coefficient, were a stainless-steel regulating rod through the center of the core and three boron-carbide shim-safety rods.

Active region of the reactor was a cylinder 33.3 in. dia, 35.8 in. high, with 7.4 in. thickness of BeO reflector. Critical mass was 14.9 kg U^{235}; total mass of U^{235} in the entire system—core and external piping—was about 66 kg; flow velocity was 1 meter/sec. Designed to operate at 1.5 Mw, the reactor experiment actually ran at 2.5 Mw peak output. Inlet fuel temperature was about 1,200° F and outlet from reactor—which glowed a dull red—1,500° F.

ARE went critical Nov. 3, 1954, when molten UF_4 concentrate was added to the barren mixture of molten fluoride salts, and was brought to full power Nov. 9. It ran continuously for four days and four nights; after logging some 90 Mw-hr of high-power, high-temperature operation, it was shut down. Thereafter it ran critical considerably longer at full temperature and pressure but *not* full power, and was eventually dismantled. Cost of the machine alone, exclusive of development costs, was about $3-million. Weinberg would not say why operation at power was so brief, contenting himself with the remark, "The trick isn't to make these things critical, but to make them run at high power." It was clear that all further work on fused-salt systems since 1954 has been for the account of the aircraft program and is still within the clammy confines of classification.



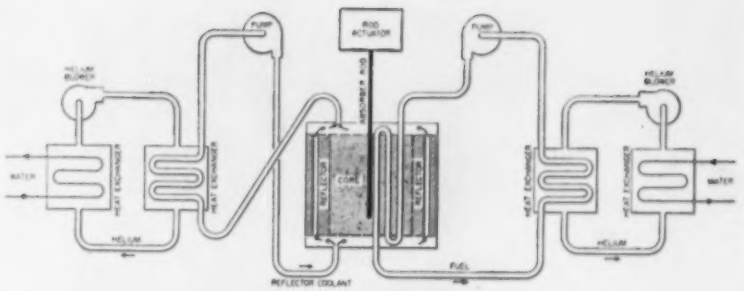
A.R.E. CORE: Reactor vessel (below) with thermocouple leads attached to bottom. Top view of core (above) shows serpentine nickel-alloy fuel tubes emerging from hexagonal BeO moderator elements

Significance for Civilian Power

But ARE did demonstrate some things of pertinence to the civilian power reactor field, and these Weinberg enumerated:

- "Again the extreme stability of a liquid fuel reactor was noted." ARE was completely power-demand responsive, and control was done mainly by manipulating the flow of cooling helium to the heat exchanger; control rods were used mainly to set operating temperature level.
- Xenon-135 came off continuously along with many other gaseous fission products.
- It proved possible to prevent freeze-up by keeping pipes always heated.
- "As in HRT, maintaining absolute leak tightness was difficult. It was solved adequately for the relatively short period of operation of ARE; however, there was a very small leak from the gas volume above the liquid surface of the fuel pumps which made the operation awkward during part of the time."

"Altogether," the Oak Ridge chief concluded, "the experience with ARE reinforces our view that chemical-type reactor systems merit serious investigation, that even a high-melting salt mixture can be used as a reactor fuel, and that one way to achieve economic nuclear power—perhaps the simplest way—is to burn U^{235} in very high-temperature devices like ARE in which the fuel is liquid. . . . The incentive for achieving a successful high-temperature reactor that burns U^{235} in liquid form is seen from the following economic consideration. If U^{235} costs $17/gm, and if it can be burned without refabrication (i.e. in liquid form) in a very high-temperature reactor with a thermal efficiency of 35%, the fuel cost, just as a straight burner, amounts to less than 2 mills/kwh. This is competitive with the best conventional fuel costs. The trick is that the system

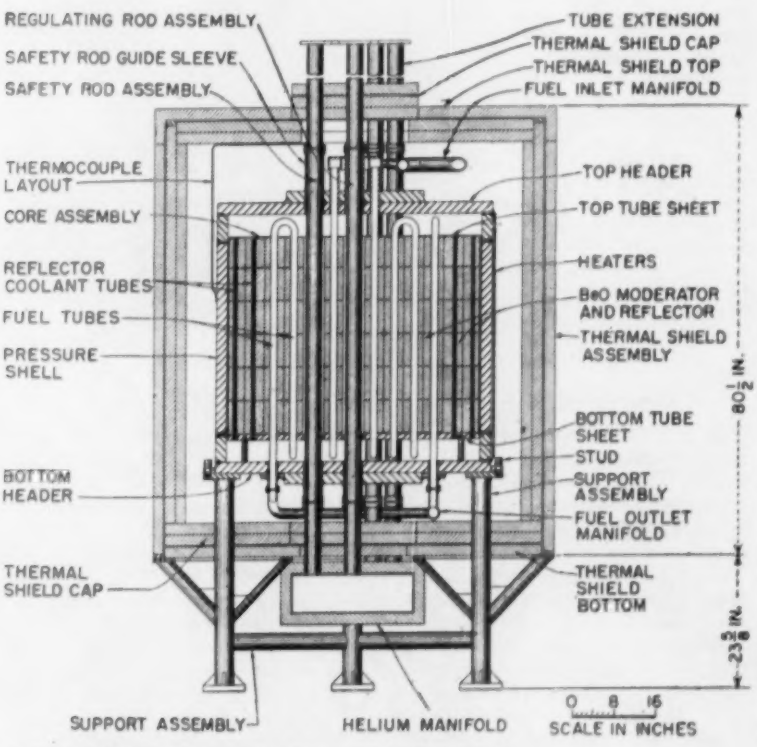


FLOW CHART for ARE shows core-cooling and reflector-cooling loops

must use fuel in fluid form because if it were in solid form, reprocessing costs would make the fuel cost much greater than 2 mills."

High thermal efficiencies comparable with the best modern conventional plants (35–40%) can be attained, he indicated. The question of whether corrosion in fluoride systems can be held down sufficiently to allow a molten fluoride reactor to operate for many years is one on which ORNL is now working, Weinberg said. The fused salt system in many ways resembles LMFR, he continued, but fluorides have one great advantage over bismuth: it is impossible to carry enough thorium in solution in bismuth to make a blanket, whereas a fluoride breeder blanket material is available: ThF_4–Li^7F–BeF_2. Disadvantages of fluorides are their high melting point —932° F; they are not as good moderators as heavy water and therefore require a greater U^{235} concentration; and, because the fluoride salts used as carriers absorb some neutrons, breeding ratios achievable are less favorable than in D_2O systems.

If the work being done by a study group at Oak Ridge continues to develop favorably, Weinberg said, it is planned eventually to build another reactor experiment to demonstrate the "long-term reliability of components and compatibility of materials" for application to molten fluoride central station power plants.



ELEVATION SKETCH of reactor shows construction details

Direct-Cycle Takes Lead For Nuclear Plane Engine

The Air Force has put half the $200-million left of its Aircraft Nuclear Propulsion funds for next year into General Electric's X211 direct-cycle project, according to *Aviation Week*, a McGraw-Hill publication. Pratt & Whitney conversely has had its share cut, due to complications—probably heat transfer—met in its indirect-cycle engine project, J91. The J91 is described as a large, low-temperature—hence low-performance—engine. One nuclear consulting firm is reported to have said that all its aircraft program business is now with GE. Following successful operation of a turbojet engine by reactor heat at the Idaho reactor testing station (NU, Aug. '56, R1; March '57, R6), GE has transferred some of its top men from its jet engine departments to the X211 project at Evendale, Ohio, and has also increased its staff in Idaho, where running of experimental engine configurations is continuing. The first Idaho test is reported to have involved four J73 turbojets with combustors removed, hooked up to a direct-cycle air-cooled, water-moderated reactor.

Difference between the GE direct and P&W indirect cycles (sketches NU, Aug. '56, 75) is this: GE flows the turbojet's compressor discharge air directly through ducting in and around the reactor to pick up the heat; P&W flows compressor discharge air through an intermediate heat exchanger to pick up reactor heat from a liquid (probably liquid sodium) primary loop. Basic problem with the latter is that even subsonic turbojets need 1,200° F air going into their turbines, which means the heat exchanger tube walls must be at even higher temperatures. For supersonic engines, where air temperatures of 2,000° F entering turbines are required, the heat exchanger materials simply are not available. And unless hotter cycles can be operated, the nuclear system cannot develop the power-to-engine-weight ratio to lift the additional load that shielding imposes.

Explanation in Congress

The foregoing helps interpret testimony by Air Force Secretary Donald A. Quarles before the House of Representatives Defense Appropriations subcommittee, explaining the A-plane stretch-out (NU, March '57, R6, May, 22). Quarles disclosed the fiscal '58 ANP appropriation ($213-million) would be less than that spent in the current

year, fiscal '57 ($237-million), and less than a third of that originally planned ($700-million). "The thing that forced [this] reconsideration," he told the subcommittee, "was the fact that after we laid down the program and contemplated pushing ahead on aircraft prototype as well as the prototype of two different kinds of nuclear-propulsion engines for a bomber, our engineering estimates from our airframe contractors and from our propulsion contractors kept getting more and more unfavorable. We got to the point where, in order to accomplish these things, they estimated we had to build a plane that would weigh considerably in excess of our specifications." The first A-plane that could now be put in the air would be "a flying reactor" and not a useful military weapon, he said. "There was a time when we had a program to produce such a flying reactor without too much regard to its military usefulness. We abandoned that program on the ground it was a luxury we could not afford. It would not contribute to our military potential and therefore we would not attempt to do it. . . . [As for] reverting to a program twice this in size, which was the program we had a year ago, it would not be in the national interest to spend money of that kind."

Venezuela, GE Sign Pool Reactor Contract

A $3.6-million contract has been signed by General Electric and the Venezuelan Institute of Neurology and Brain Research covering a 3–5-Mw (heat closed-pool-type research reactor (NU, Nov. '56, R1). The unit, to be built at GE's San Jose plant, is scheduled for delivery in April 1958, with installation to be completed by December. It will be set up at the Institute's hilltop hideaway at Altos de Pipe, 7 mi outside Caracas, at 5,000 ft altitude.

Designed by Walter Zinn's General Nuclear Engineering Co., it is described as Latin America's most versatile reactor. It is designed especially for research in medicine, agriculture and industry; will have both horizontal and vertical tubes, two thermal columns, and a thermal neutron flux of 1.6×10^{13}. It will be fueled by 20% enriched, U. S.-supplied uranium. Buried in the ground, it will have a spheroid steel-and-concrete containment dome 80 ft dia and 70 ft high; flanking it will be two wings, one for medical research, the other for basic research in nuclear physics, chemical and petrochemical applications. Enclosed in three protective shields of steel and concrete 13 ft thick, the reactor will have as an additional safety feature four radio-controlled seismograph outposts surrounding the site to shut down the reactor automatically in case of earthquake.

Dr. Humberto Fernandez-Moran, director of Venezuela's national nuclear program, last month told the Inter-American Symposium at Brookhaven that the Institute had already invested $7-million in the project and plans a total investment of about $50-million.

Phillips Makes Offer

Phillips Petroleum has offered the Venezuelan government technical assistance on atomic energy as part of an oil concession package proposal. Phillips would help in obtaining radioisotopes and in nuclear research and development, supply six full-time experts for one year.

English hole . . .

On May 1, this was appearance of No. 1 Reactor at Berkeley-on-Severn, where ground had been broken Jan. 7 for one of Britain's first two commercial nuclear power plants. AEI–John Thompson is building two-reactor, 300-Mw total, station. In pit center will stand 1,000-ton pressure vessel, surrounded by eight heat exchanges to be erected in indentations seen at pit circumference. One large crane has been assembled, three-legged tower for another is underway

. . . American hole

Excavation is being completed 47 mi SW of Chicago for Commonwealth Edison's 180-Mw Dresden station. Some 400 men are preparing base for 190-ft steel sphere. Rectangular cut in foreground will become fuel-handling canal; saucer-shaped hole in background will accommodate sphere. Chicago Bridge & Iron Co., who will build sphere, are preparing to move in now; plates are to start arriving in August, and sphere is to be completed next spring. Reactor vessel is to arrive on site in mid-'58

Germany Forging Rapidly Ahead —

Three developments attest to Germany's accelerating pace in nuclear power:

1. Buys Reactor at Home

Close on the heels of the announcement of a first private German power reactor purchase—an AMF Atomics boiling reactor, by RWE of Essen (NU, May '57, p. 27)—came word of a second. And it came as swift corroboration of gloomy prophecies by some American atomic industry people that a U. S. reactor market in industrialized areas abroad would last only until the foreign countries could build their own; and that Germany in particular would soon be a stiff competitor. Not only did this second contract for a German nuclear power plant go to two German firms, but it embodies a novel reactor concept.

This is a solid-homogeneous type, gas-cooled and graphite-moderated. Fuel elements are ball-shaped, of sintered uranium carbide and graphite. They are fed into a tower-like reactor tank from above, and unloaded below. Thus the reactor will be controlled merely by changing the loading and the coolant flow; there will be no safety or control rods. A high rate of burnup is hoped for by resintering the elements at intermediate intervals, thus curing radiation-damaged elements. The concept permits operating temperatures considerably higher than in any existing reactor: 980° F steam will be produced. This is held in turn to promise power costs competitive with coal-fired power plants in Germany.

Contract for the design of a 15-Mw net (electrical) pilot plant went to Brown Boveri & Co. of Mannheim (German subsidiary of Brown Boveri of Switzerland) and Friedrich Krupp of Essen. Site has not yet been chosen, but a 1960 target date for operation of the pilot plant has been set. Placing the order is AVR, a six-city utility group (NU, Feb. '57, R9) now grown to nine cities in northwest Germany—Düsseldorf, Duisburg, Hagen, Minden, Bremen, Hameln, Hannover, Kiel and Wuppertal.

2. Triples Power Goal

A tripling of Germany's nuclear power goals was announced last month by Atomics Minister Siegfried Balke—the first program enlargement by one of the six Euratom countries since the Armand-Etzel-Giordani report was published last month (*see p. 25*).

Balke said he would soon propose a ten-year, $510-million program of nuclear energy investment that would provide nuclear capacity of 1.5-million kw in West Germany, as opposed to 0.5-million by 1965 in the present program. He said power demand would soon exceed the capacity of conventional power methods, but that he felt 1.5-million kw of nuclear capacity was the most Germany can build in the next 10 years.

This is 10% of the total Euratom goal of 15-million kw by 1967. In Europe it has been assumed that West Germany's share would have to amount to 40% if the goal is to be met.

Balke's program is based on the assumption the power reactors to be built will be largely based on natural uranium, supply of which is being arranged (NU, May '57, 27). In addition to the 500 tons natural uranium from Canada and 1,500 kg enriched from the U. S., Germany is also counting on imports of natural U from France and some limited deposits in Bavaria.

3. Centrifuges U Cheaply

Another new German success in the search for a cheaper method of separating U^{235} is reported to NUCLEONICS from Bonn. Last year Prof. E. W. Becker at the University of Marburg was reported making progress with his separation-nozzles or -jets (NU, June '56, 19A).

Now Prof. Wilhelm Groth of Bonn University has developed a method of using the centrifuge process that was tried at Oak Ridge during the Manhattan District days and found wanting.

Groth uses uranium hexafluoride gas, led continuously through two centrifuges operating at 40,000 rpm; their rotating 0.8-in. thick steel cylinders are about 6 in. high, 2-in. i.d. He claims to have obtained enrichments up to 2%. He will get nine larger centrifuges installed in his Institute at Bonn this fall; and in addition, the state of North Rhine–Westphalia plans to establish no fewer than 50 centrifuges at its new nuclear research center at Düren (*see p. 27*). This plant, Groth says, will produce 5 tons/yr of enriched uranium, at power costs one-eighth to one-tenth those of diffusion plants. No extra electric power plants will be required.

Russian Goal for 1960 Is 2-Million Kw Electrical

One of the minor mysteries about Russia's nuclear power program was cleared up last month: the goal of 2–2.5-million kw by 1960, set at the 20th Party Congress in January 1956 is electrical—not thermal—output.

Americans who study Russian pronouncements, newspapers, and periodicals have felt a certain degree of equivocality about the nature of the Russian goal. Since the original account in *Pravda* of the original Party Congress speech by Igor Kurchatov, one of Russia's leading nuclear scientists—which quoted him referring specifically to electrical output—the only Soviet references have been by the vaguer term *obshchaya moshchnost*, "total power." Other indications seemed to perpetuate the confusion, as pointed out in RAND report P-1076 of May 1 by Arnold Kramish. Direct inquiries by NUCLEONICS to Kurchatov and to Efim P. Slavsky, director of the Russian AEC, brought no replies.

Last month another Soviet nuclear scientist, in the U. S. with the International Atomic Energy Agency preparatory commission, V. S. Emelianov, told NUCLEONICS that the Russian target is definitely, in kilowatts of electrical output.

Two More U. S. Firms Take Partners Abroad

Two new partnership arrangements completed last month by U. S. companies with foreign firms for cooperation in the nuclear field illustrate a definite trend toward cross-licensing arrangements as an alternative to straight sales of complete reactors. Atomics International reached an agreement for "extensive cooperation" with Sweden's huge ASEA electric power equipment company. Foster Wheeler teamed with Vereinigte Kesselwerke (United Boilerworks) of Düsseldorf, West Germany. Two earlier such agreements were between AMF Atomics and Mitchell Engineering of London (NU, March '56, 24), and Alco Products and Humphreys & Glasgow Ltd., also London (NU, Feb. '57, R4).

Other U. S. arrangements completed or in preparation will team Chicago Bridge & Iron Co. with Motherwell Bridge & Engineering of Britain, and Westinghouse with Siemens-Schuckert of Germany. In addition, Westinghouse and General Electric have their world-wide international organizations.

U. S. Reactor Salesmen Sell—or Irk—Japanese

American atomic energy experts at the joint conference of the U. S. and Japanese Atomic Industrial Forums held in Tokyo, Osaka and Nagoya last month took the opportunity to do some nuclear reactor sales promotion.

The basic American theme expressed at the conference—which brought together 89 Americans and more than 1,450 Japanese and other delegates from 14 observer Asian nations (plus 200,000 at the exhibit)—was the superiority of U. S. reactors over Britain's Calder Hall.

This sales pitch was probably effective to some degree, for the advantages of American reactors were spelled out for many Japanese experts for the first time. However, it is unlikely that Japan will, as a result, abandon her plans to purchase a Calder-type reactor. But the chances are better now—indeed, good—that this country will also buy an American reactor.

Still, the Americans' heavy-handed sales approach is being criticized in some quarters. Some Japanese as well as American representatives believe the U. S. could have made its point more subtly.

British Miffed

British observers in Tokyo were indignant, declaring that British efforts to sell Calder Hall here were never accompanied by derogatory references about U. S. reactors. Especially resented there were remarks by W. Kenneth Davis, director of AEC's division of reactor development, in view of the fact that he is a U. S. government official. Davis had emphasized that Calder Hall is "inherently very large and expensive to build."

Other Americans said Britain has not yet built a single-purpose reactor, pointing out that Calder Hall produces electricity as a by-product in the manufacture of plutonium. It is held, on the other hand, that U. S. reactors—both pressurized water and boiling water types—are "better for Japan" because: (1) the enriched uranium they use is a superior fuel; (2) other fuels are produced as by-products; (3) they are smaller in size; (4) because of this they are more suitable for use in nuclear-propelled ships.

While impressed with these arguments, many Japanese representatives privately maintained that they prefer to purchase a Calder-type first, because: (1) it is already producing electricity; (2) it is basically simple in design; (3) the natural uranium it uses may be preferable to enriched uranium, as Japan would be completely dependent on the U. S. for supplies of the latter.

Japan to Build Own

Actually, the question of U. S. vs British reactors lost some of its meaning with unexpectedly determined statements by Japanese speakers to the effect that Japan is rushing plans to produce her own reactor. Japan's ultimate goal is "complete domestic production" of nuclear reactors, said Chikara Kurata of the Japan Machinery Federation. Emphasizing that a Japanese model is now being designed, he added, "We wish that the U. S. favor us with technical information, conclude technical agreements, and train our atomic industrial engineers in all phases of nuclear energy techniques."

Other Japanese delegates explained privately that Japan will purchase one Calder Hall reactor and maybe one American unit, but would try to get manufacturers' blueprints at the same time so she could produce her own versions of these reactors until a domestically-designed model could be turned out—probably in a decade.

Indeed, one reason why America's atomic stock has gone up in Japan recently is that U. S. firms appear more willing than British reactor manufacturers to conclude licensing agreements with Japanese companies—as witness the pending accord between Westinghouse and the Mitsubishi group for local manufacture of pressurized water reactors (NU, April '57, 17).

Large Imports Seen

However, some Japanese government officials believe it will be necessary to import several Calder Hall units and perhaps as many American reactors in the next decade, with an increasing number of parts made in Japan.

Japanese delegates stressed that in many cases Japan needs atomic power generation "almost immediately" in view of the severe shortage of energy resources expected in the next few years. Japan, they said, must produce 1-million kw of nuclear-generated power by 1965, and 10-million by 1975. About $25-million will be spent this year on atomic energy development, it was reported.

It was said that Japan will develop nuclear-powered ships parallel with attempts to generate atomic electricity, in order to maintain her role as the world's leading shipbuilder.

Americans reported extensively on U. S. atomic development. Drawing special attention were reports on fusion reactor research by Frederic de Hoffmann of General Atomic; the possibilities of studying the human aging process, by Charles L. Dunham, director of AEC's biology and medicine division; and problems in the production of nuclear-propelled aircraft, by Alan H. Lazar of Babcock & Wilcox.

Nuclear dedication in a little valley

General Electric's $10-million nuclear research center at Vallecitos (Spanish for "little valley") near Pleasanton, Calif., was dedicated to peacetime atomic development and to reducing costs of nuclear power. Three major components of facility are: (right) radioactive materials lab, equipped for intensive study of irradiated materials; (left center) experimental physics lab, to house a critical assembly for mocking up various types of reactor cores, a nuclear test reactor (NTR) for measurement of nuclear properties and for calibrations, and a core assembly room; and (far left) 5-Mw developmental boiling water reactor, in capsule-shaped containment shell, to prove out GE's design for 180-Mw Dresden Station and to test new designs and features of boiling reactors. It will supply steam by year's end to a 5-Mw turbine generator installed by Pacific Gas & Electric for its grid. At this site also GE will build a testing reactor (see p. 26)

PWR, Calder, Boiling Costs About Alike, Euratom Finds

There is little or nothing to choose today between the cost of power from British and American reactors. This is one of the most striking conclusions arrived at by the Armand-Etzel-Giordani ("three wise men") study group in a painstaking analysis carried out for Euratom. The report, "Target for Euratom," was issued May 7 (NU, May '57, 20). The audacious Euratom goal, 15-million kw by 1967, would cost about $6.4-billion to install, the report reveals—$4-billion more than the same amount of conventional power capacity, although operating cost at 80% load factor would be only 11–14 mills/kwh for nuclear power as against 10.6–12 mills for conventional. Of the total cost $1-billion would be fuel inventory.

To get this program rolling quickly, the report says, Euratom countries must either buy reactors from the U. S. or U. K., or build them under license. The only full-scale designs immediately available are pressurized water, boiling water, and Calder gas-cooled types. Comparing these three in the order named, the report finds net fuel cost (including fabrication, reprocessing, and U-Pu credits) to be 2.7, 3.3, and 1.7 mills/kwh respectively; fuel inventory charges 1.3, 0.8, and 0.7 mills/kwh; capital charges 6.3, 5.4, and 7.4. After adding other operating costs and 25% contingency, the cost range is calculated to be 11.3–14.1 for pressurized water, 10.5–13.1 for boiling water, and 10.8–13.6 for Calder-type.

The report also comments that in the circumstances of Euratom, "it is extremely advantageous to be able to get U^{235} at the very favorable terms announced by AEC. . . . There is no basis for apprehension about the employment of enriched fuel on the ground that AEC prices are subsidized and therefore exert an artificial and unreliable effect on design."

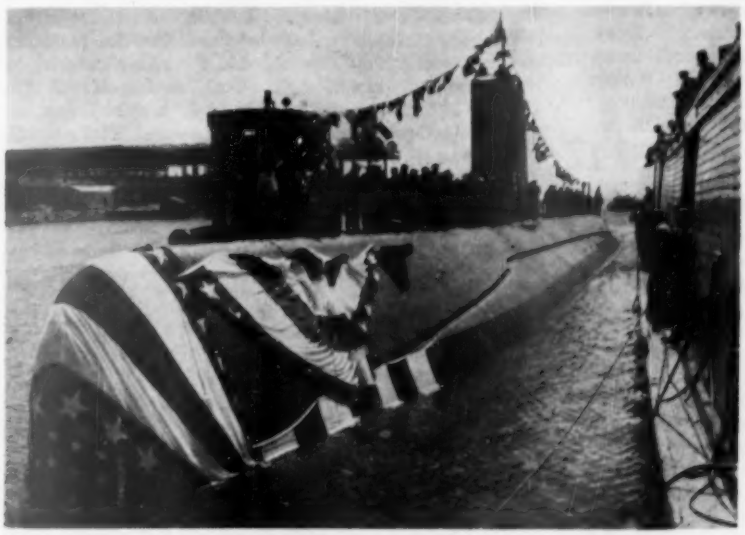
AEC Licensing News

Construction permit issued: The Martin Co., to build nuclear experiment facility at Middle River, Md., 5/13/57

Export licenses issued: ACF Industries, to export 5-Mw pool-type research reactor to Italy, 5/16/57

AMF Atomics, to ship 1-Mw pool-type reactors to Greek AEC, Athens, 5/17/57; and to McMaster Univ., Hamilton, Ont., 5/17/57

General Electric, to export a 3-Mw pool-type reactor to Junta de Energia Nuclear, Madrid, 5/22/57



A-sub—off-the-shelf model

A few minutes after becoming water-borne, Skate—the Navy's (and world's) third nuclear-powered vessel and first 'assembly-line' A-sub—is towed to her fitting-out dock. Launching on May 16 marked passing of nuclear submarine construction program from developmental phase to quantity production: three sistership of Skate are already on navy-yard ways, two at Portsmouth, N. H., one at Mare Island, Calif. With almost the same hull silhouette as Nautilus, Skate displaces 2,400 tons, about 25% less than either Nautilus or Seawolf, and her 265-ft length is about 60 ft less than either; she will be slower, as her Westinghouse pressurized water reactor, half the size of Nautilus', has only half the output. Navy Undersecretary Franke said at launching that 'Skate soon will be considered the conventional attack submarine'

Texas Isotope Spill, BNL Blast Mar Month

Two "atomic incidents" marred last month's news.

• On May 2 it was learned that 50 days earlier, on March 13, in the lab of the M. W. Kellogg Co. nuclear products division at South Houston, Texas, irradiated pellets of a mixture of compressed iridium-192 and aluminum metal powder had been broken while being uncanned. "Severe contamination of the laboratory"—in AEC's words—resulted. Nevertheless the lab remained in use 29 more days, the extent of contamination apparently not being realized; and it was 37 days before Kellogg notified AEC on April 19. The firm had been encapsulating Ir^{192} in radiographic cameras it sold to test welds and castings.

The container, from Idaho, was being opened in an open-top hot cell with a master-slave manipulated cutting tool, which accidentally cut into the pellets. These disintegrated and a cloud of radioactive (low-level, 75-day half-life) dust particles spread through the lab. Four men present were exposed, but received only small doses; later three additional men were found to be exposed, as well as homes, cars, a pet dog. It was Kellogg's second radiation-burn incident in two years.

On May 2, 13 days after it learned of the incident, AEC ordered a suspension of Kellogg's isotope license. Tracerlab was brought in to survey and decontaminate the area. Kellogg fired its plant manager. On May 11 AEC amended its radioisotope license regulation to require immediate notification of "potentially serious accidents." Kellogg says it had decided months earlier to close out its nuclear products division as uneconomic. AEC gave the firm's construction division permission to resume using sealed sources in its own work.

• After 10 years' operation with a clean record, Brookhaven National Laboratory suffered the supreme irony of having an accident while it was playing host to 100 distinguished Latin American nuclear scientists.

On May 15 an unexplained explosion took place in apparatus for pilot-plant study of fuel recovery by volatility methods using bromine trifluoride; unirradiated natural uranium was used in this test. The violence of the blast blew off a steel shield, permitting the corrosive halogens to splash out on four workers. Three were treated and sent home, the fourth severely burned, is recovering.

Reactor News

ARGENTINA HAS A-PLANT OFFER

The Argentine government is studying an offer to build an $80-million, 280-Mw nuclear power station for the city of Buenos Aires, NUCLEONICS has learned; Burns & Roe of New York would do the engineering on the project in cooperation with the Argentine Technical Institute for Light and Power. The offer was first presented last July, and is still under consideration. It was presented by the J. Christensen Financial Corp., a firm with Argentine officers incorporated in New York in July 1956 to promote investment of U. S. capital in Argentina. Type of reactor or fuel would be decided in a preliminary study; the plant would be built in 30 months. Christensen would operate it at no profit for a 15-year amortization period, selling power to the government at a price not to exceed the mean cost of conventional power, after which the plant would be turned over to the government.

CANADA REVISING NPD DESIGN

The possibility of having pressure tubes rather than one large pressure vessel in Canada's first power reactor is one of the major design changes contemplated that has resulted in a temporary stoppage of all work at the Des Joachims, Ont., site of the 20-Mw Nuclear Power Demonstration reactor (NU, Oct. '56, 28). While excavation down to solid rock and general site preparation had been underway, Atomic Energy of Canada at Chalk River, 20 mi downstream on the Ottawa River, was completing a study for a 200-Mw reactor. Designs proposed for the larger reactor brought into focus some questions about the NPD design: it was felt the smaller reactor should incorporate and test advanced techniques for its successors. The next few months accordingly will be spent on revision of the NPD design.

FIRST REPORT GOOD ON CERAMIC FUEL

Four Argonne reactor engineers last month gave the American Ceramic Society annual meeting a first report on behavior of the urania-thoria ceramic core now in Borax-4 (NU, March '57, R8). The pioneering fuel elements consist of a mixture of uranium oxide and thorium oxide pellets sintered together and assembled in an aluminum high-nickel jacket. The core was designed to test those properties of ceramic reactor fuels that make them so promising: high corrosion resistance, ability to withstand high temperatures, and low irradiation-creep rate. Initial tests demonstrate good resistance to corrosion in both water and NaK; and, operating under more extreme heat transfer conditions than previous Borax cores, the ceramic elements stood up satisfactorily at quite high levels of power output per unit volume of core.

B & W CRITICAL FACILITY BUSY

A first full-scale experimental mock-up of the core for Con Edison's Indian Point power reactor went through initial critical tests in Babcock & Wilcox's critical experiment laboratory at Lynchburg, Va. In the zero-power experiments, the Indian Point fuel elements (NU, April '57, 20) were represented by assemblies of aluminum plates to which are taped polyethylene strips containing U^{235} and thorium powder. A second concrete cell to house a second critical facility is nearing completion adjacent to the existing one (NU, June '56, 21B); it will be used to obtain nuclear physics data needed to design the pressurized water reactor for the first nuclear merchant ship (NU, May '57, 21).

WTR BACK ON TRACK . . .

After having put off ground-breaking late in December during a "re-evaluation of capital-expenditure priorities" (NU, Feb. '57, R10), Westinghouse is proceeding with construction of its materials testing reactor at Waltz Mill, Pa., 29 mi. SE of downtown Pittsburgh. WTR, for which AEC has already issued a construction permit, was the first privately-owned test reactor to be announced.

. . . AND GETR STARTING

Construction will start shortly on a $4-million materials testing reactor at General Electric's Vallecitos Atomic Laboratory, it was announced at the lab's dedication ceremonies (*see photo, p. 22*). Scheduled to be completed in late 1958, the General Electric Test Reactor will be a 30-Mw (heat) ORR or tank-in-a-pool type, with fully-enriched uranium in 20 flat-plate fuel elements. It will be enclosed in a capsule-shaped containment vessel 66 ft dia, 82 ft high (70 ft above grade, 12 ft below), similar to the one containing the Vallecitos developmental boiling water power reactor. Irradiation of test samples will be possible both inside the reactor's aluminum pressure vessel and outside it in the pool. Initially, GETR will be used in developing nuclear fuels.

AMF DUTCH REACTOR CRITICAL

The pool research reactor AMF designed and, with the help of Dutch industry, built for the Netherlands became the first reactor sold abroad by U. S. private industry to go critical, on May 20 (NU, Oct. '56, R5; March '57, R12). The reactor will operate at 10 kw until Sept. 16 as main feature of the international exhibit "Het Atoom" (The Atom) at Schiphol Airport, Amsterdam; thereafter it will be moved to the Netherlands Nuclear Research Center for 100-kw operation.

PRECRITICAL TESTS AT OMRE

Early last month precritical testing of the Organic Moderated Reactor Experiment began at the Idaho reactor testing station under the direction of Charles A. Trilling, OMRE project chief for Atomics International. AI is building the unit for AEC. Construction was essentially completed late in April; the reactor is slated to go critical this fall.

SPERT SPECS ISSUED

Plans and specifications for the $1.3-million construction contract for SPERT-2 were issued late last month. This second of the Special Power Excursion Reactor Tests (NU, March '56, 46; June '56, 46) is a low-pressure (300 psi), low-temperature (400° F) system to test effects of different moderators and reflectors (light and heavy water, graphite, beryllium) on transient behavior and safety. The work, expected to take 15 months, includes a 60- × 92-ft reactor building; construction of non-nuclear portions of the reactor system including reactor vessel, process piping and coolant system; electrical power system; waste disposal system; and three storage tanks of 14,000, 15,000 and 20,000 gals.

World News

New Site Chosen for German Labs

After long deliberation, the state government of North Rhine–Westphalia has abandoned Königsforst just west of Cologne as site for its big nuclear research center (NU, July '56, R1; Oct. '56, R10; Jan. '57, R9). Instead the $20-million center will be located at Düren, halfway between Cologne and Aachen, and somewhat more remote from large population centers. The lab complex, comprising 12 institutes, is to be completed in four years; half its cost is to be paid by the Federal Republic. It will serve primarily the training and research needs of four nearby universities: Cologne, Bonn, Münster, and the technical university at Aachen. Two British research reactors—a Merlin pool and a Dido heavy-water materials testing reactor—were ordered last summer.

British Briefs

Sir Christopher Hinton prophesied that nuclear power stations will be producing power by 1980 at half the cost in conventional stations; by 2000 even domestic space heating would come primarily from nuclear power, he said. . . . F. J. Erroll, parliamentary secretary to the Board of Trade, told Commons the Atomic Energy Authority is considering bismuth for future reactor systems but has no present plans for stockpiling the metal. He said it must "remain unknown for some time" what type of reactor will be chosen for eventual "large-scale production." "LMFR systems are still in an early stage of study—not even of development," he said; "there are many competing forms of reactor awaiting development." He was explaining why the government rejected a proposal by Mining and Chemical Products Ltd. to stockpile bismuth. . . . A super-size nuclear power station of 400–500 Mw was proposed by Central Electricity Authority for the Hinkley Point, Somerset, site (NU, Dec. '56, R1). The English Electric–Babcock & Wilcox–Taylor Woodrow consortium is negotiating on this third CEA station, which would have two graphite-moderated, gas-cooled (improved Calder type) reactors. . . . Negotiations are under way for formation of another large nuclear energy development organization between the airplane builder Hawker Siddeley and John Brown, the Clydebank shipyard that built *Queen Mary* and *Queen Elizabeth*. Hawker Siddeley had already formed a nuclear power subsidiary (NU, Dec. 56, R11). . . . Orders for nuclear power plants are expected to top $1.26-billion by 1958. Of this, one-third will be for export, probably to Japan, Italy, Germany and Holland. To meet demand for small—10–20 Mw—stations, Britain may obtain purchase rights to U. S. power plants until new British designs become available. Meanwhile British gas-cooled plants are being designed capable of burning slightly-enriched uranium at outputs as low as 60 Mw. (Calder's gross output is 92 Mw.)

French Nuclear Navy School Set

Certain that nuclear energy will power the ships of the future, the French Navy is establishing a school for atomic marine propulsion at Cherbourg, to train naval officers at all levels: officers, petty officers, engineers and sailors. Apprenticeships will be given at the national nuclear center at Saclay, where five naval officers are already studying nuclear naval engineering. An embryo crew for the first A-ship, the submarine *Q-244* (NU, May '57, 27) is being chosen.

Radiation Study Center for Japan

A $4.7-million atomic research center for study of fallout, radiation effects on the body, treatment of radioactivity-induced diseases, and training of doctors and technicians, is to be built near Tokyo at Tokai village, site of Japan's Atomic Energy Research Institute and her first research reactor. The new lab, to be completed in 1960, will be called the National Basic Radiotherapeutic Research Institute; it will have 256,000-ft² total floor space sprawled over 50 acres, and will make Tokai village the radiotherapeutic center of Japan, if not of the entire Orient.

Saskatchewan Uranium Booming

Northern Saskatchewan's Beaverlodge is shaping up this spring into a $60-million/yr uranium field accounting for 22% of Canada's total contract production. Seven companies are shipping, or preparing to ship, ore to three concentrator mills in the area with a combined capacity of 4,100 tons/day, expected to rise to 4,600 tons by midsummer. Several other companies are awaiting completion of spring thaw to step up geophysical, drilling and underground exploratory work.

Czechs Hard Put to Meet Russian Orders

"The Lenin Works of Pilsen [the former Skoda Works] are unable to fill Soviet delivery orders for nuclear machinery and gear on time," said Czech Deputy Premier Karel Polacek. Last year the Lenin Works exported only 69.4% of the required volume, causing a $7.7-million shortage in this state enterprise export business. Moscow had ordered machinery and gear, tools, and secondary material for Soviet nuclear plants early in 1956. Czechoslovakia was forced to change the Lenin Works' traditional field—machines for metal and wood working—to switch to the atomic machinery the Soviet demanded. Polacek stressed that "Czechoslovakia's own atomic power plant will also get machinery from the Pilsen firm—after the export obligations have been fulfilled."

Belgium Eyes Congo Reactor Market

The newly-formed Société Belgonucléaire, interested in installing nuclear power plants in the Belgian Congo, has made a study showing that boiling water reactors with slightly-enriched cores in sizes from 1 to 10 Mw (electrical) would be competitive today in about half of the Congo, in areas out of transmission reach of existing hydro plants.

Internuclear Gets Italian Tie

Internuclear Co., nuclear consultants of Clayton, Mo., has been chosen consulting engineers for design and construction of a large power reactor in southern Italy planned by SENN—Società Elettronucleare Nazionale (NU, April '57, 27) [not to be confused with SELNI—Societa Elettronucleare Italiana (NU, Oct. '56, R10)]. Reactor type, site, etc., remain to be decided.

AI Ships Water-Boiler to Denmark

Twenty-five crates containing a 5-watt water boiler research reactor were in the cargo of the Swedish express freighter *Seattle* when she sailed from Los Angeles for Copenhagen. The Atomics International unit, fueled with uranyl sulfate solution, is for the Danish nuclear research center on Risø peninsula 20 mi W of the capital, and is to be in operation this summer.

News in Brief

Report on HRT

The Homogeneous Reactor Test ran at full temperature and pressure for over 1200 hr early this year, Oak Ridge National Laboratory disclosed; natural uranium was circulated under full design conditions for 500 hr, and equipment performance was excellent, except for high-pressure feed pumps and letdown valves, now being improved. At present all high-pressure system flanges are being replaced, because of chloride contamination (NU, Jan. '57, R4); criticality may be possible by Fall. ORNL also revealed it is planning a 150-Mw (heat), 6×10^{14} flux homogeneous research reactor and a 50-Mw thorium breeder HRE-3.

Nautilus Heads for Pacific

Nautilus' second core "represents a significant advance" over her original fuel charge that drove her 69,138 mi in 26 months, Westinghouse discloses. Propelled by her new and more efficient core, the pioneer A-sub is getting her first taste of Pacific waters: she sailed May 21 from Panama for Pearl Harbor, Seattle, San Francisco and San Diego, is due back at Panama July 14 and at New London late in summer.

Second Reactor for N. C. State

A second research reactor will be built by North Carolina State College, owner of the country's first private university reactor. The new unit, to be completed next summer, may be an Argonaut or a pool type, and is to operate at 50–100 kw; Daystrom, Cook Electric, American-Standard and General Nuclear Engineering Corp. have submitted bids. It will be housed in the building now accommodating the first water-boiler, using the latter's shielding and instrumentation. The water-boiler, which has been operating for a month following a two-year shutdown due to core leaks (NU, March '57, R9), will be moved to new quarters.

Lithium Tails Available to Industry

Foote Mineral Co. has revealed that it is one of three producers supplying AEC with lithium; under a five-year contract, it supplies "virgin" lithium hydroxide. In spite of the easing of security under which this information was released, the quantity of material involved and AEC's end use for lithium are still classified. It was stated, however, that AEC is interested in Li^6, one of the two naturally-occurring isotopes. Li^6, only 7% of virgin lithium, has a high neutron-absorption cross section and has been used not only in manufacture of H-bombs but also in reactor shielding. In addition it has potential as a tritium breeder in the blankets of fusion reactors. Foote has an option on lithium "tails"—material depleted in Li^6—remaining of virgin LiOH it supplied to AEC, and this is available for commercial use. Low neutron cross section of Li^7 makes these tailings attractive as starting material for lithium reactor coolant. Foote says that in spite of its contract with AEC, "there is more lithium available to industry than there has ever been."

New Slate at NEWA

Clare M. Cotton Jr. of the *Wall Street Journal* was elected president of the Nuclear Energy Writers' Association, succeeding NUCLEONICS' editor, Jerome D. Luntz, who had led the organization in its first two years. Corbin Allardice of the World Bank and Roger P. Hammond of General Electric are new vice-presidents, and Grace Wells Urrows, public information chief for AEC in New York, the new secretary. Edwin A. Wiggin of the Atomic Industrial Forum continues as treasurer.

AEC-Forum-ANS Set Safety Meeting

AEC, the Atomic Industrial Forum, and the American Nuclear Society will co-sponsor a technical conference on reactor safety in New York Oct. 31. An unclassified meeting, it follows the Forum-ANS conferences.

Atlantic Research a Nuclear Newcomer

A program of consulting and project services will be offered by Atlantic Research Corp. of Alexandria, Va., recent entry in the nuclear engineering field. Michael Markels, Jr., previously with Columbia Univ.'s heat transfer facility, directs similar operations at Atlantic.

J. J. Hopkins Dead; Pace Succeeds

In 1937 John Jay Hopkins became a director of Electric Boat Co., a small shipyard pre-eminent in its specialty: submarines. In 1946, he started expanding with acquisition of Canadair, a major Canadian plane builder; in 1952, on becoming board chairman as well as president, he set up General Dynamics Corp., with Electric Boat, its Electro Dynamic Co. subsidiary, and Canadair as divisions. Convair and Stromberg-Carlson Co. followed by the merger route, and in 1955 the General Atomic division was created. Five of six GD divisions are heavily involved in nucleonics. In 1956 GD became one of about 30 U. S. firms to chalk up net sales of more than $1-billion. Hopkins was active also in promoting schemes for international cooperation based on peaceful harnessing of atomic energy. Last month Hopkins, 63, died in Washington, of cancer. Executive vice-president Frank Pace, Jr., former U. S. Budget Bureau director and Secretary of the Army, succeeded as president; Earl D. Johnson, senior vice-president for operations and fiscal affairs, succeeds to Pace's post. Board chairmanship is still vacant.

AEC Signs Two Uranium Contracts

What is described as the nation's largest uranium processing mill will be constructed 25 mi north of Bluewater, N. M., by Kermac Nuclear Fuels. Costing $16-million, the mill is scheduled to have 3,300 tons/day capacity and is contracted to run until Dec. 31, 1966 . . . Homestake-Sapin, a partnership formed by Homestake Mining and Sabre-Pinon, will construct a $9-million uranium mill in the Ambrosia Lake area of New Mexico. The mill is committed to a $117-million contract for processing 1500 tons/day capacity.

Wah Chang Dedicates Zr Plant

A privately-financed zirconium sponge plant at Albany, Ore., built in six months and operating since January, was dedicated by Wah Chang Corp. of New York. Wah Chang has also been operating the Bureau of Mines' Zr plant, also at Albany, for AEC since May 1956; combined production of the two plants makes Wah Chang the world's largest Zr producer, the firm says. It all now goes to AEC, but by July Wah Chang hopes to have reactor and commercial grade Zr available for direct sale to private industry.

Glenn Martin Shortens Name

The Glenn L. Martin Co. at a stockholders' meeting voted to shorten the title to The Martin Co.

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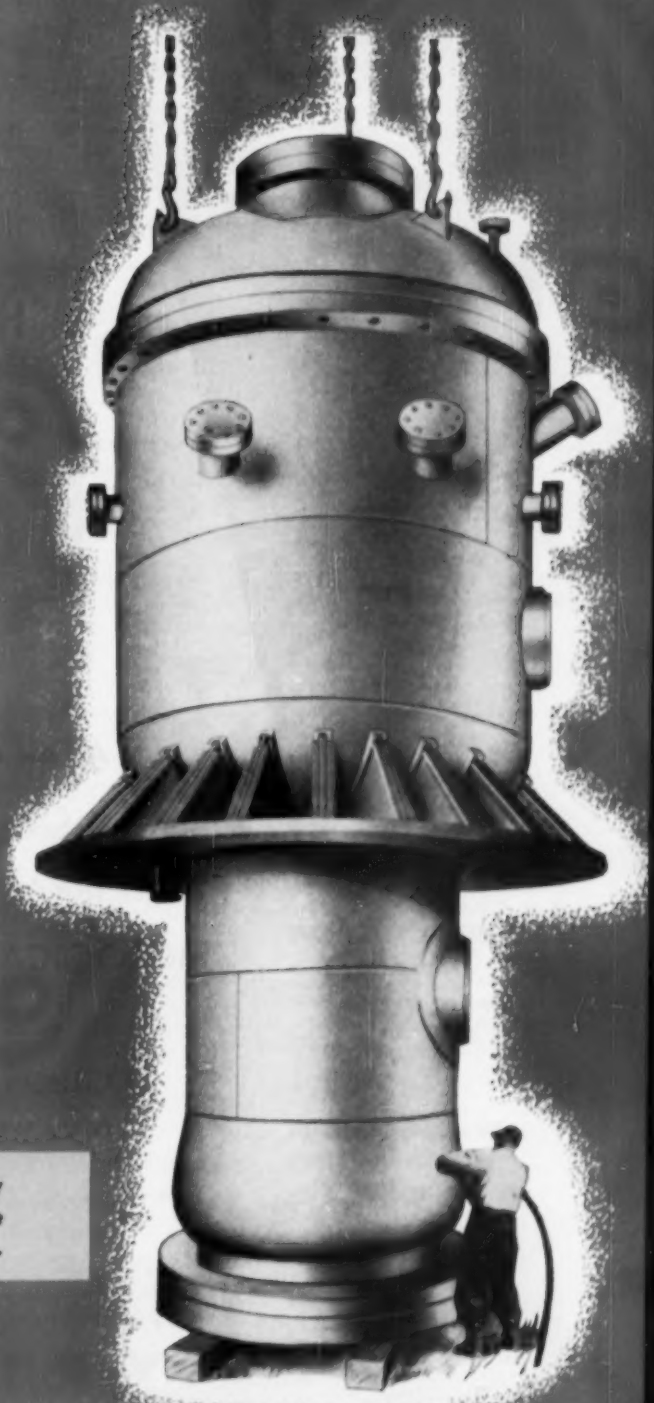

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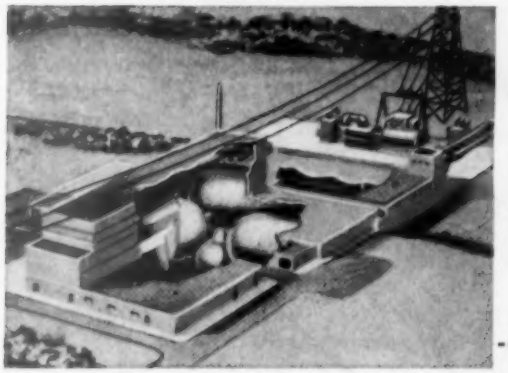

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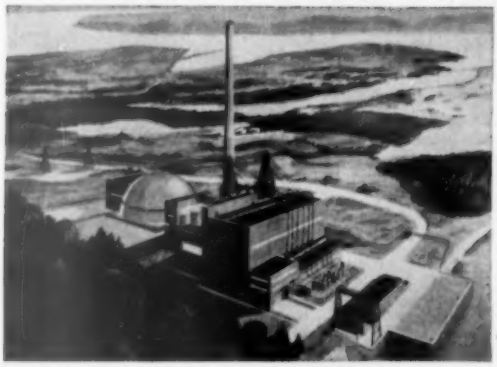
of the AEC and Duquesne Light Company

*Steam Generators. Two of the four steam generators for America's first full scale commercial atomic power plant have been designed and fabricated by Foster Wheeler. They will convert heat from the pressurized water reactor into steam for operating the turbo generator.

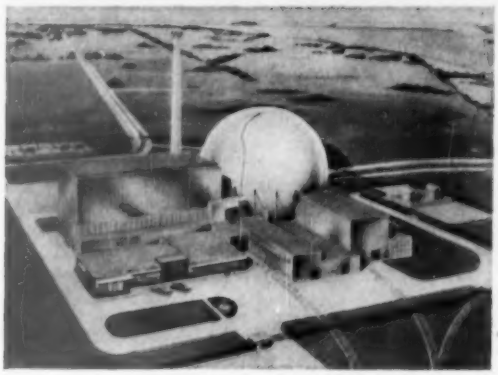
*Pressurizer. Also supplied by Foster Wheeler is the 18-ft high, 300 cu ft pressurizer used to maintain coolant loop pressure. It is designed for 2500 psig at 675 F and contains 342 electric immersion heaters for controlling operating pressure.



275,000 KW INDIAN POINT PLANT

of the Consolidated Edison Company

*Separately-Fired Superheaters. Saturated steam from the pressurized water reactor will be superheated from 447F to 1000F by two Foster Wheeler oil-fired superheaters, each with a capacity of 1,075,000 lb/hr at 410 psia. Compared to operation with saturated steam direct from the reactor, addition of the superheat cycle will raise plant capacity from 163,000 kw to 275,000 kw and reduce estimated plant cost from $350 to $255 per kw.



180,000 KW DRESDEN STATION

of the Commonwealth Edison Company

*Primary Steam Separator. Primary steam generated in the reactor will be separated from the water in a Foster Wheeler steam drum designed for a pressure of 1250 psia and capable of delivering 1,400,000 lb/hr at 1000 psia with a moisture content of not over 0.1%.

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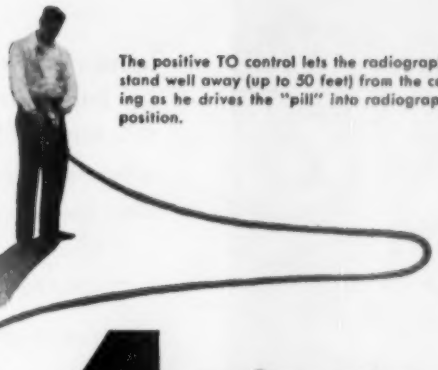

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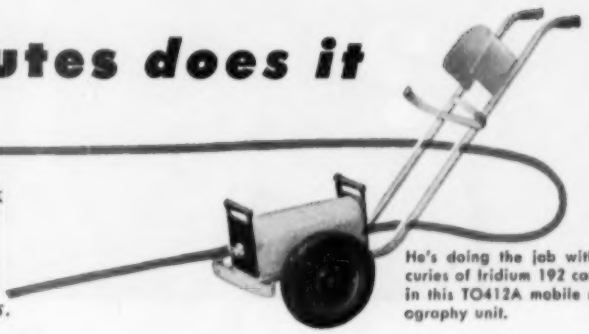

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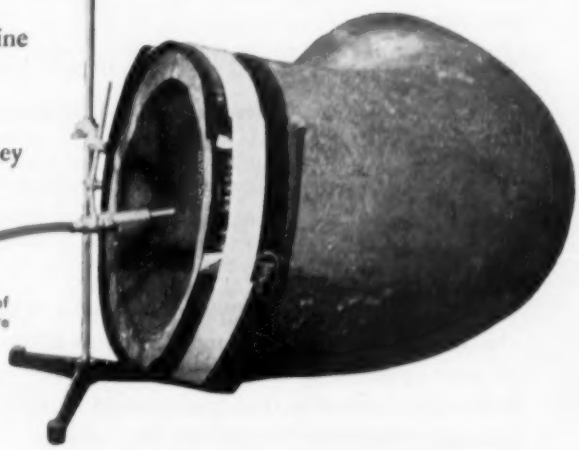

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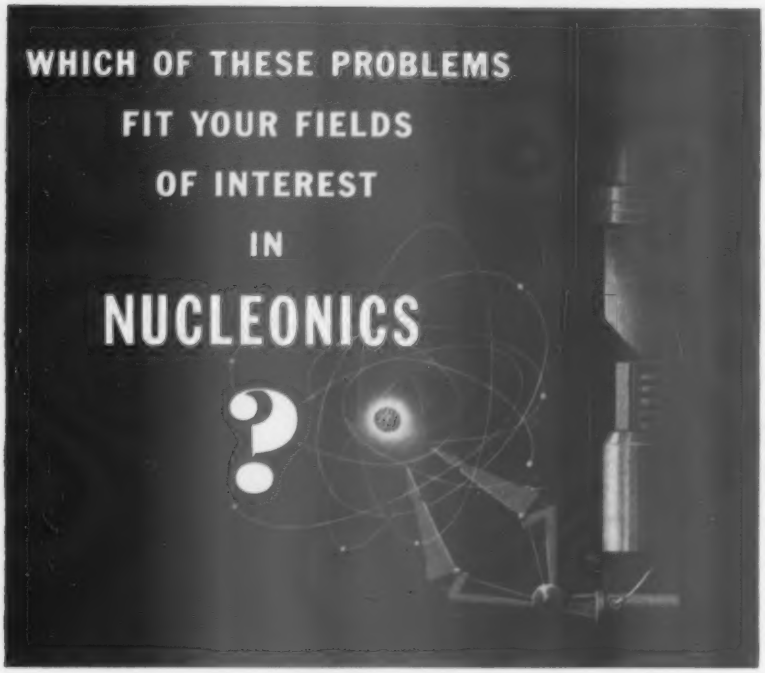

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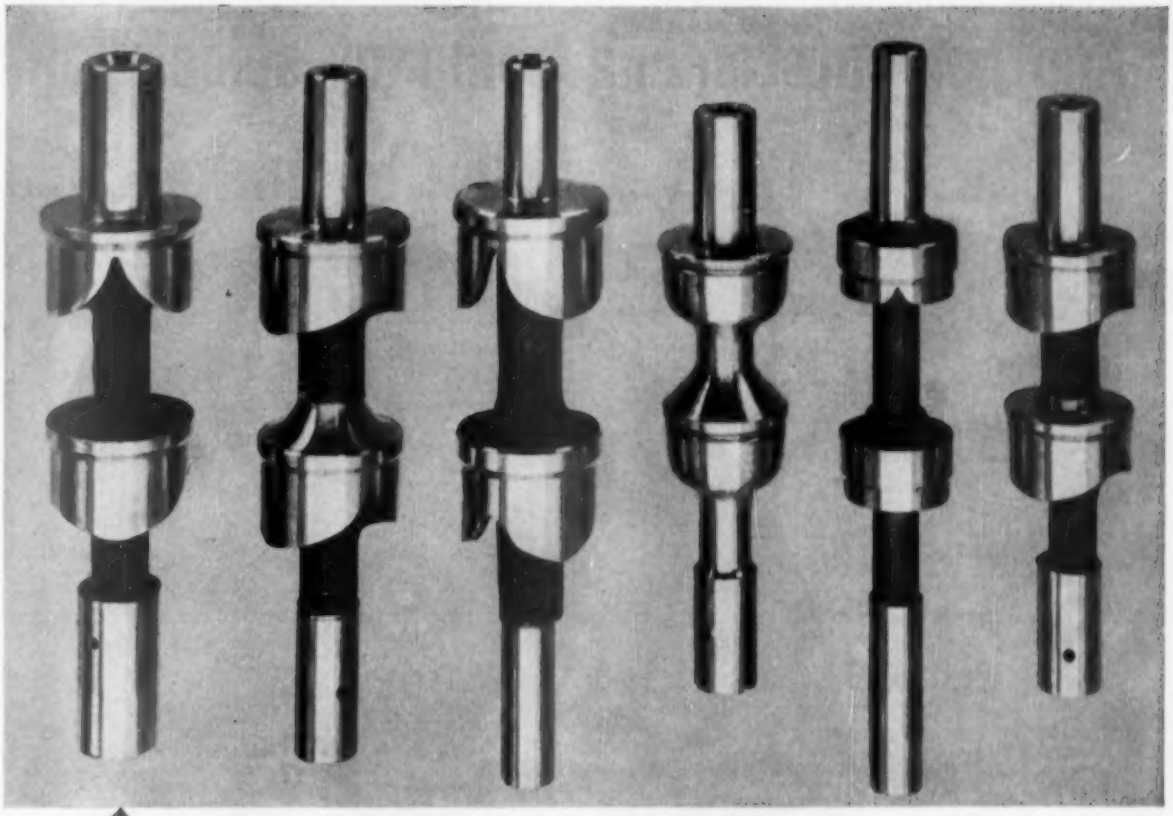

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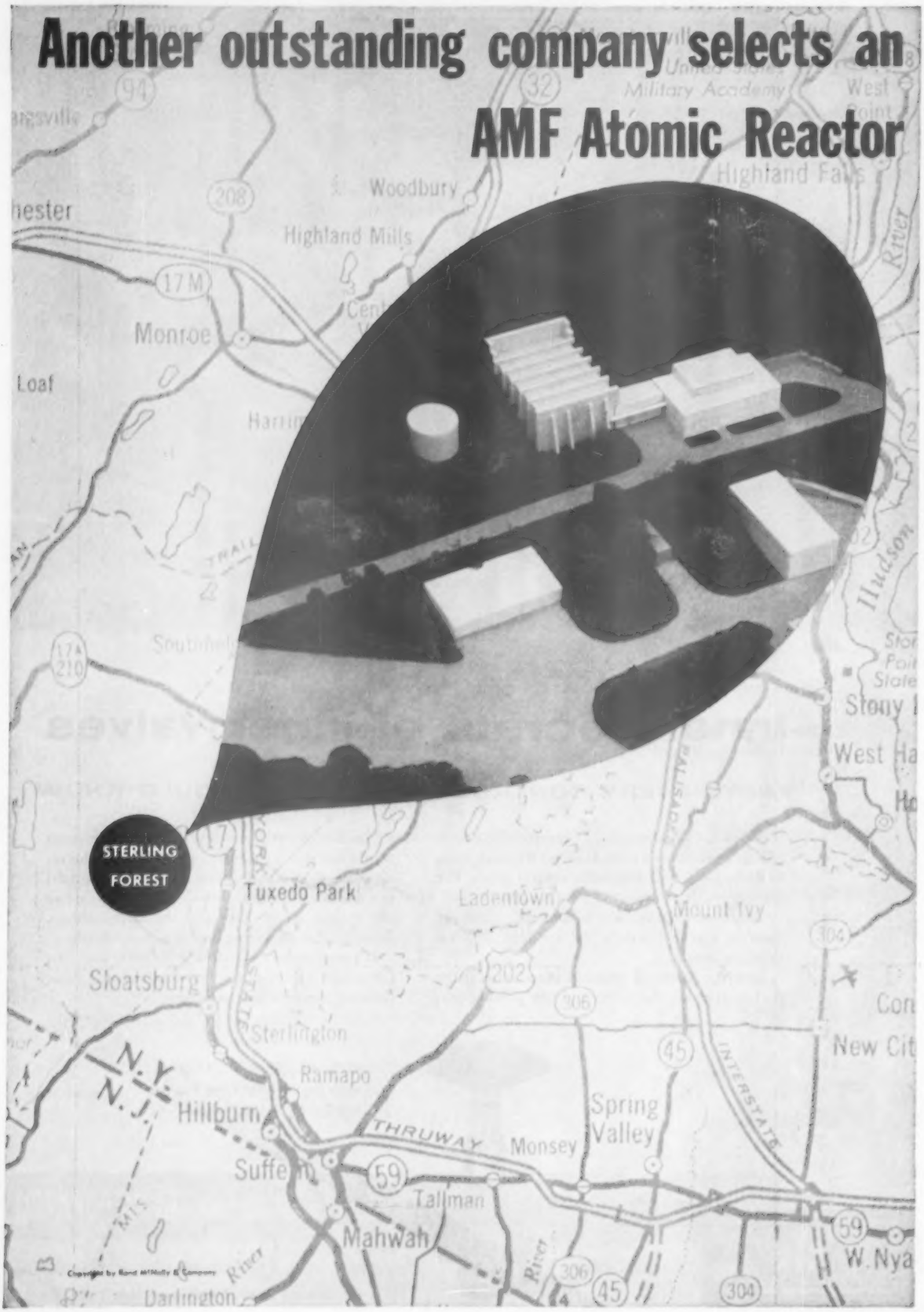
Another outstanding company selects an AMF Atomic Reactor



5 megawatt pool type research reactor to be built for **Union Carbide Corporation**

This new AMF reactor—to be located in Sterling Forest, just outside Tuxedo Park, N. Y.—brings to a total of *12* the research and power reactors either completed, under construction, or under contract by AMF ATOMICS.

A larger version of the pool type reactor now operating at Battelle Memorial Institute, the new reactor will be the heart of Union Carbide's new Nuclear Research Center. It will be devoted to the commercial application of nuclear energy in plastics, metals, gases, chemicals, and carbons. This project is in addition to Union Carbide's activities at Oak Ridge National Laboratory which the corporation operates for the U. S. Atomic Energy Commission.

Like all AMF-designed reactors, this new reactor is based upon AMF's famous "unitized" engineering principle, a form of "building block" design which provides not only for economical construction, but for great reactor flexibility as well.

Average neutron flux in the order of 3×10^{13} n/sq cm-sec. will be available. Controls are of the thoroughly-proved AMF design now in use in reactors throughout the world, including those soon to go "critical" in Germany and Holland.

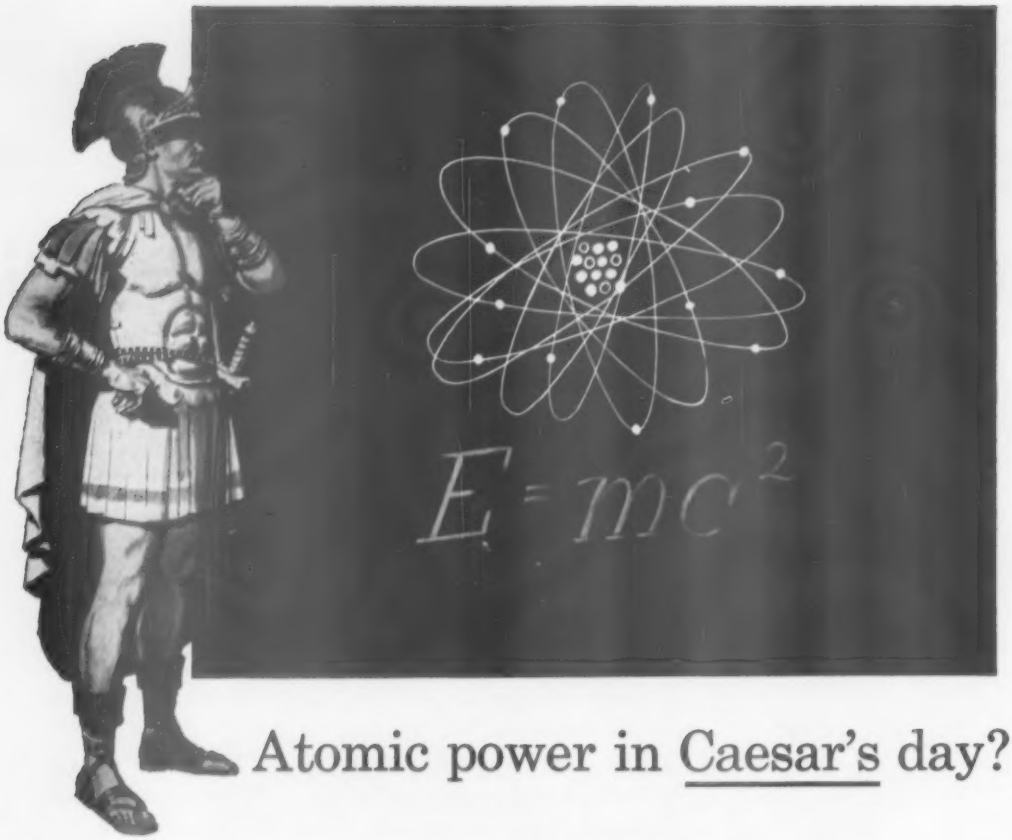
In addition to the safety features inherent in this type of reactor, a number of other AMF-developed safety features have been incorporated, adding still further to the reliability and safety of the reactor. Facilities will include two 8" and four 6" diameter beam tubes, a 4' x 4' thermal column, and a number of hydraulic and pneumatic "shuttles", in addition to the access to radiation provided by the pool proper.

If *you* have a reactor project in mind, or are considering the purchase of associated handling or control equipment, follow the lead of an atomic pioneer. Discuss your plans with AMF ATOMICS. You'll find the world's leading designer and producer of research reactors an invaluable partner. For more information on AMF reactor designs, including pool type, light and heavy water tank-type, educational training reactors, and closed-cycle boiling water power reactors, simply write the address below.

AMF ATOMICS

AMF ATOMICS
Division of AMERICAN MACHINE & FOUNDRY COMPANY
AMF Building • 261 Madison Ave. • New York 16, N. Y.

- *Power and Research Reactors*
- *Radiation Process Equipment*
- *Reactor Control Systems*
- *Radio-active Material Handling Equipment*



Atomic power in Caesar's day?

Certainly!

It was there, in the ground, in the air and water. It always had been. There are no more "raw materials" today than there were when Rome ruled the world.

The only thing new is knowledge . . . knowledge of how to get at and rearrange raw materials. Every invention of modern times was "available" to Rameses, Caesar, Charlemagne.

In this sense, then, we have available *today* in existing raw materials the inventions that can make our lives longer, happier, and inconceivably easier. We need only *knowledge* to bring them into reality.

Could there possibly be a better argument for the strengthening of our *sources* of knowledge—our colleges and universities? Can we possibly deny that the welfare, progress—indeed the very *fate*—of our nation depends on the quality of knowledge generated and transmitted by these institutions of higher learning?

It is almost unbelievable that a society such as ours, which has profited so vastly from an accelerated accumulation of knowledge, should allow anything to threaten the wellsprings of our learning.

Yet this is the case

The crisis that confronts our colleges today threatens to weaken seriously their ability to produce the kind of graduates who can assimilate and carry forward our rich heritage of learning.

The crisis is composed of several elements: a salary scale that is driving away from teaching the kind of mind *most qualified* to teach; overcrowded classrooms; and a mounting pressure for enrollment that will *double* by 1967.

In a very real sense our personal and national progress depends on our colleges. They *must* have our aid.

Help the colleges or universities of your choice. Help them plan for stronger faculties and expansion. The returns will be greater than you think.

If you want to know what the college crisis means to you, write for a free booklet to: HIGHER EDUCATION, Box 36, Times Square Station, New York 36, New York.

HIGHER EDUCATION

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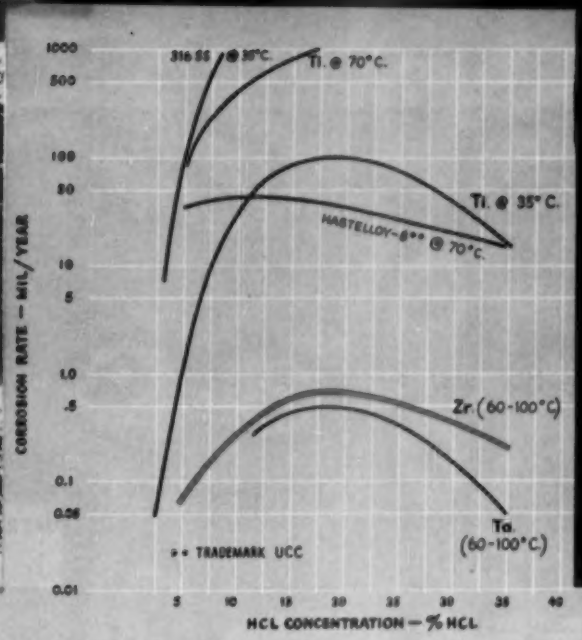

want to keep a "hot fluid" in its place?

C-N ZIRCONIUM

Now, you can look at your product or process with zirconium in mind! For example, tests indicate that only tantalum is comparable to zirconium in resisting hydrochloric acid — and zirconium is less costly.

C-N Zirconium will be available in commercial quantities in the near future. Now is the time to investigate the remarkable corrosion resistance of zirconium—as demonstrated in the nuclear field and in certain industrial applications.

Columbia-National is undertaking a product research program to investigate the merits of zirconium in specific media. If you have severe corrosion problems, C-N offers laboratory and research facilities to establish how and where zirconium can solve your problem. Write us about your requirements. Send for technical data on properties and corrosion resistance of this new metal.

NEW METALS FOR INDUSTRY

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Jointly owned by
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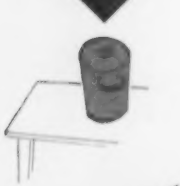
CG-1

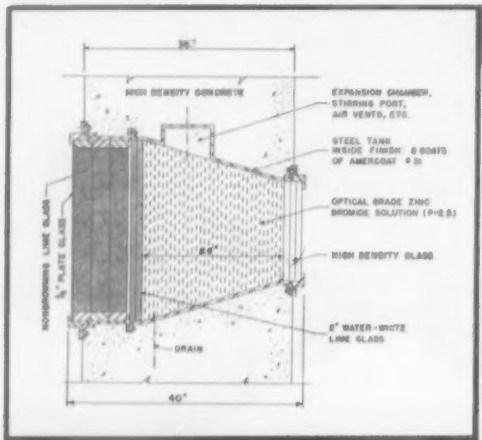
Radiation Shielding by RAY PROOF®

- *PIONEERS IN THE CREATIVE USE OF MATERIALS FOR STRUCTURAL RADIATION SHIELDING*
- *JUDGE RAY PROOF BY THE CUSTOMERS IT KEEPS*

SHIELDING from a capsule to a building



nuclear shield
x-ray protec
light proo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OPTICAL GRADE ZINC BROMIDE SOLUTION
HIGH DENSITY GLASS
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40"

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Hamilton Standard
Div. of United Aircraft Corp.
Pratt & Whitney Aircraft
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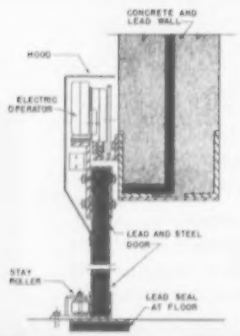
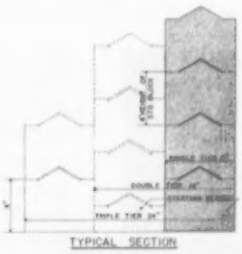
Westinghouse Electric Corp.



SHIELDED & MOTORIZED RADIATION DOORS



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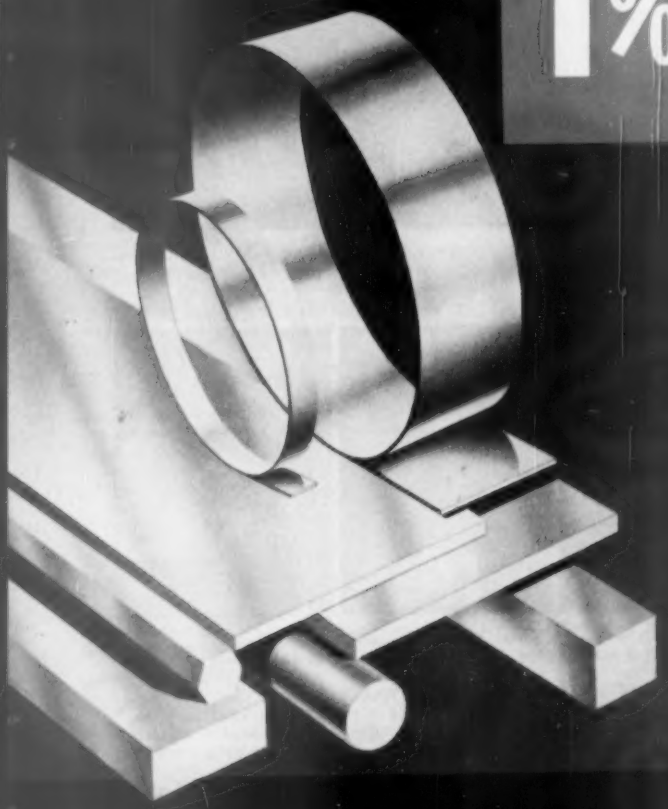
1% stainless



zirconium
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and other
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1% Boron Stainless, with its remarkable thermal neutron absorption property, was developed by and is now available from Superior Steel in strip, sheet, plate, bars and fabricated forms.

Superior's knowledge of and specialization in rolling zirconium, titanium, uranium and other difficult metals provides high precision in meeting your requirements for these new metals.

Your inquiries addressed to our Special Products Division are invited.

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Available on request—a new 1% Boron Stainless Steel research bulletin. Ask on your letterhead for Bulletin No. 57-N.

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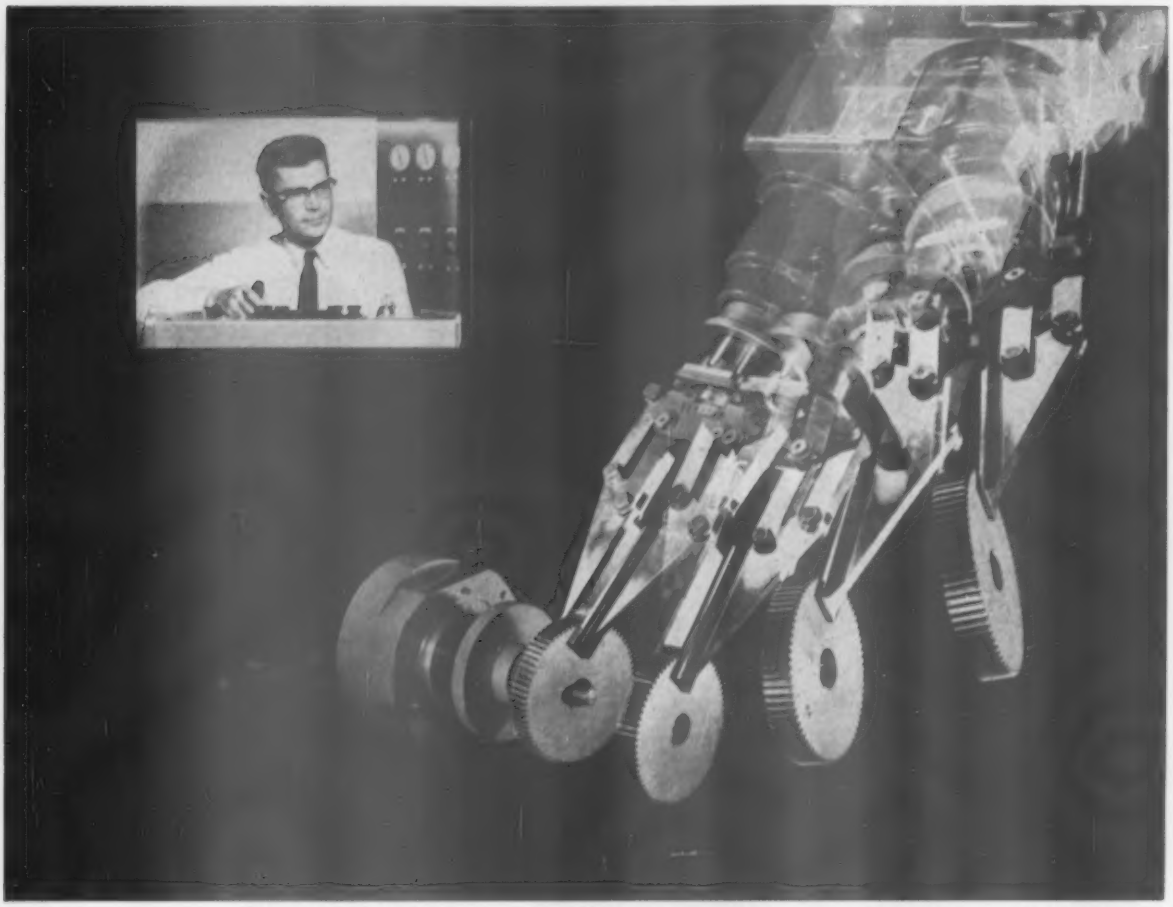
CARNEGIE, PENNSYLVANIA

This versatile "muscle-man" handles delicate or heavy duty equipment with equal ease



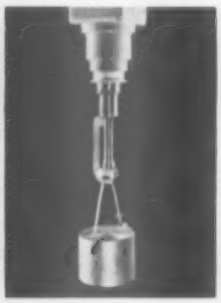
Heavy-duty and delicate operations within a hot lab are adroitly performed with the General Mills Mechanical Arm. For special requirements, telescopic or television viewing gives the versatile device nearly limitless "reach."

General Mills Mechanical Arm
performs hazardous human tasks remotely.

This easily controlled electro-mechanical robot magnifies many times the strength and dexterity of its human operator in radioactive, volatile, toxic or other lethal areas. It lifts and carries 750-pound loads, but also handles delicate laboratory equipment with like agility. Unlike the human arm, its wrists can turn continuously, and it will maintain any position or gripping force for unlimited periods. Adaptable to any installation requirements, it can be mounted on a crane or vehicle to provide complete, remote mobility.

GET MORE FACTS—SEND FOR BOOKLET
Write: Dept. N-6, Mechanical Division, General Mills, 1620 Central Ave. N.E., Minneapolis 13, Minn.

"Finger" grip may be varied from a soft, delicate squeeze to a clamp force of 150 pounds. Numerous special grasping devices are available.

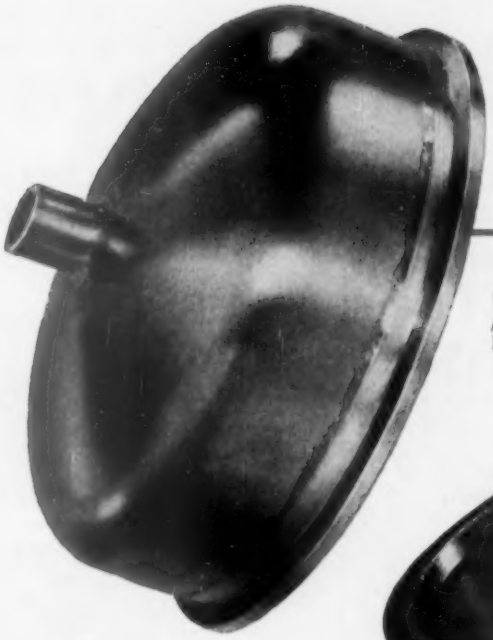


This "hook" is used for carrying and lifting or for holding some types of tools. It and the jaws are easily interchangeable remotely.

Coordinated pistol-grip handles permit a natural movement which quickly becomes instinctive. Visual-audio signals indicate grip force.

MECHANICAL DIVISION OF General Mills

CREATIVE RESEARCH AND DEVELOPMENT + PRECISION ENGINEERING AND PRODUCTION



ZR CORROSION RATE

Less than .0002 inches per year in *boiling, concentrated* hydrochloric acid.

World's First Commercial Zirconium Reaction Tank—another *first* with ZIRCONIUM by The CARBORUNDUM METALS Company. Reaction Tank manufactured by The Pfaudler Company.

MORE...in corrosion resistance to either acids or caustics...with

ZIRCONIUM

ZIRCONIUM GIVES YOU greater corrosion versatility than any other commercial metal. Reaction Tanks made of ZIRCONIUM, for example, provide resistance against strong mineral acids *and* strong caustics or organic compounds. Two-way resistance to corrosion not only gives added protection against contamination but it *now* permits an interchanging of basic components in chemical processing equipment. ZIRCONIUM valves, tanks, filter presses or pumps can be interchanged as desired for service in either strong acids or alkalis.

For further information on the many corrosion-resistant benefits of ZIRCONIUM, write to The CARBORUNDUM METALS Company. Ask to be placed on the mailing list of "More ZR Facts", the ZIRCONIUM industry's first continuing source of technical information.

The CARBORUNDUM METALS Company
Akron, New York
Division of THE CARBORUNDUM COMPANY

Production Pioneer of ZIRCONIUM

New Ray Stoppers

NATIONAL LEAD REPORTS RECENT DEVELOPMENTS IN LEAD SHIELDING

INGENIOUS NEW FORMS STEP UP LEAD BRICK SHIELDING UTILITY

For years lead brick has been widely used for movable shielding. Now, this low-cost, highly efficient material has even greater versatility.

Specially shaped bricks form circular walls without mortar

New, specially shaped National Lead brick lay without mortar in tight circular arcs. Inner surfaces may be contoured to the arc or its chord. Brick ends are developed on the arc radius (See diagram).

First "service" station for submarine reactors . . . made with curved National Lead brick.

Normally, as in the transition shield for submarine reactors shown, the brick is specially designed for special applications. It also finds use, however, to provide protection from wide angle sources.

New 400% harder brick resists dents, nicks

Damage from ordinary wear and tear is greatly reduced by a new National Lead brick, "Hi-Hard". "Hi-Hard" is some 400% harder (19 Brinell) than ordinary lead brick, resists the nicks, dents, and mashed corners that force premature scrapping of "soft" brick. The hardness is gained at slight expense in density (only 1.8%).

Hi-Hard Standard

Extrusion insures uniform density, aids cross-section predictability

No internal holes, no surface "pockets" in National Lead extruded brick.

Adequacy of shielding depends on a predictable, uniform density of shieldant as well as a predictable low impurity content. National Lead is producing extruded brick that provides both.

Extrusion prevents cavity formation within the brick. It also eliminates preferential solidification and surface "pockets" in which impurities might lodge. Extrusion assures uniform maximum density; smooth, clean surface and profile; easier decontamination.

Exposure radiation cell

Heavy water filter

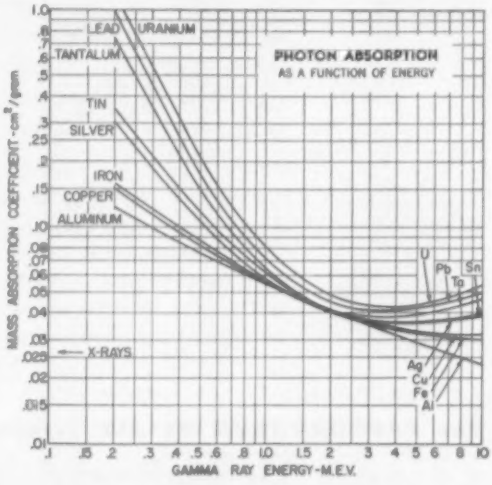
Isotope shipping container

Lead-filled steel jackets shield these new devices.

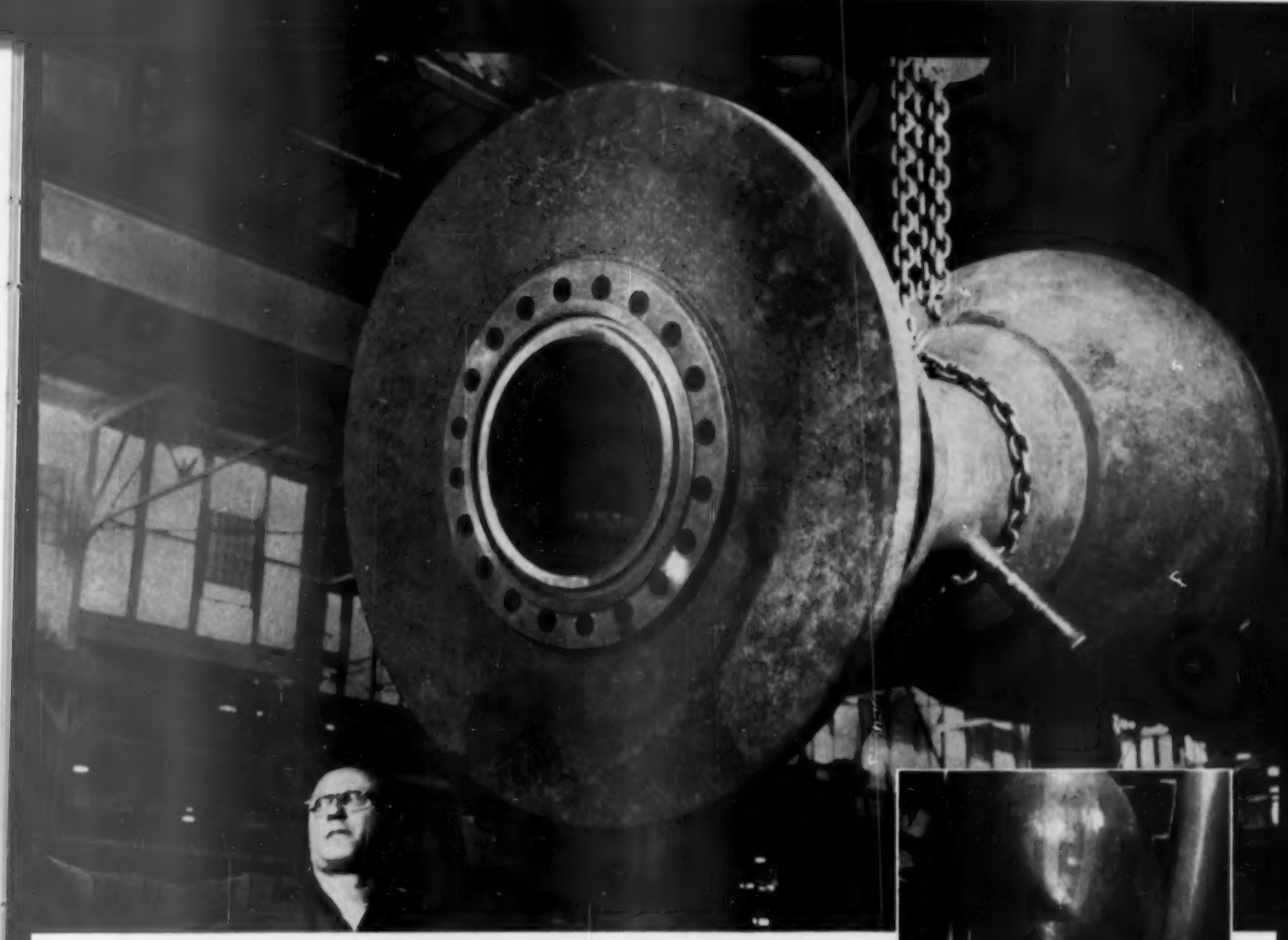
NEW VOID-FREE CASTING BOOSTS USE OF LEAD-FILLED JACKETS

Recent National Lead improvements in lead casting methods are leading to much wider use of lead-filled steel jackets for shielding containers and process vessels. With integrated jackets these devices are more compact, easier and less costly to decontaminate. The new process assures a void-free, uniform density filling.

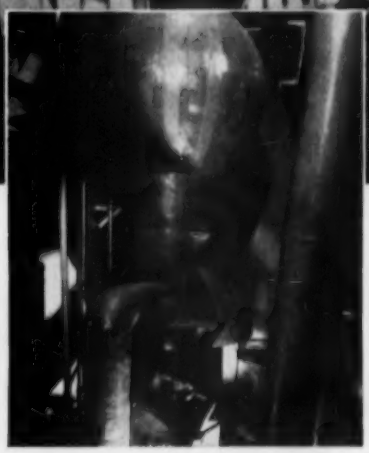
EFFECTIVENESS OF VARIOUS SHIELDING MATERIALS



NATIONAL LEAD SHIELDING, NATIONAL LEAD COMPANY, *General Offices: 111 Broadway, New York 6*



ALCO designed-and-built components for APPR-1 include pressurizer (in shop above and installed at right). ALCO built plant on prime contract from Army and AEC.



NUCLEAR COMPONENTS: ALCO HAS THE EXPERIENCE

From Manhattan Project to APPR-1, ALCO nuclear components have led in design and operating efficiency

Nuclear components from ALCO are backed by experience that began in the Manhattan Project and continued through EBR, S1W, S2W, S3G, S4G, PWR, SRE and APPR. For these reactors, ALCO built advanced heat-exchanger equipment, pressure vessels, specialized closures and other items. ALCO has also been a major supplier to AEC production installations.

ALCO NUCLEAR COMPONENTS

- reactors
- reactor vessels
- control-rod drive mechanisms
- pressurizers
- steam generators
- core support structures
- core removal and handling equipment
- regenerative and non-regenerative exchangers
- liquid metal exchangers
- decay exchangers
- special heat-transfer equipment

Research facilities, complete welding, heat transfer and mechanical labs, and the first privately owned criticality facility in industry, help ALCO continue to offer the most advanced component designs.

Consult ALCO for nuclear components of any type. Contact the nearest ALCO sales office or write Thermal Products, Dept. NC-2, P. O. Box 1065, Schenectady 1, N. Y.

ALCO

ALCO PRODUCTS, INC.

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Locomotives • Diesel Engines
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And why not?

All your eggs in one basket? . . . And why not?

It means economy and efficiency for Nuclear Systems' customers.

We can provide a complete radiation package . . .

Sources, equipment, instrumentation, training . . .

And facility design.

Applications as diverse as the irradiation of foods,

Polymerization of synthetic resins

And the synthesis of chemicals.

Interested?

NUCLEAR SYSTEMS

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Budd

another example of exciting work at los alamos...

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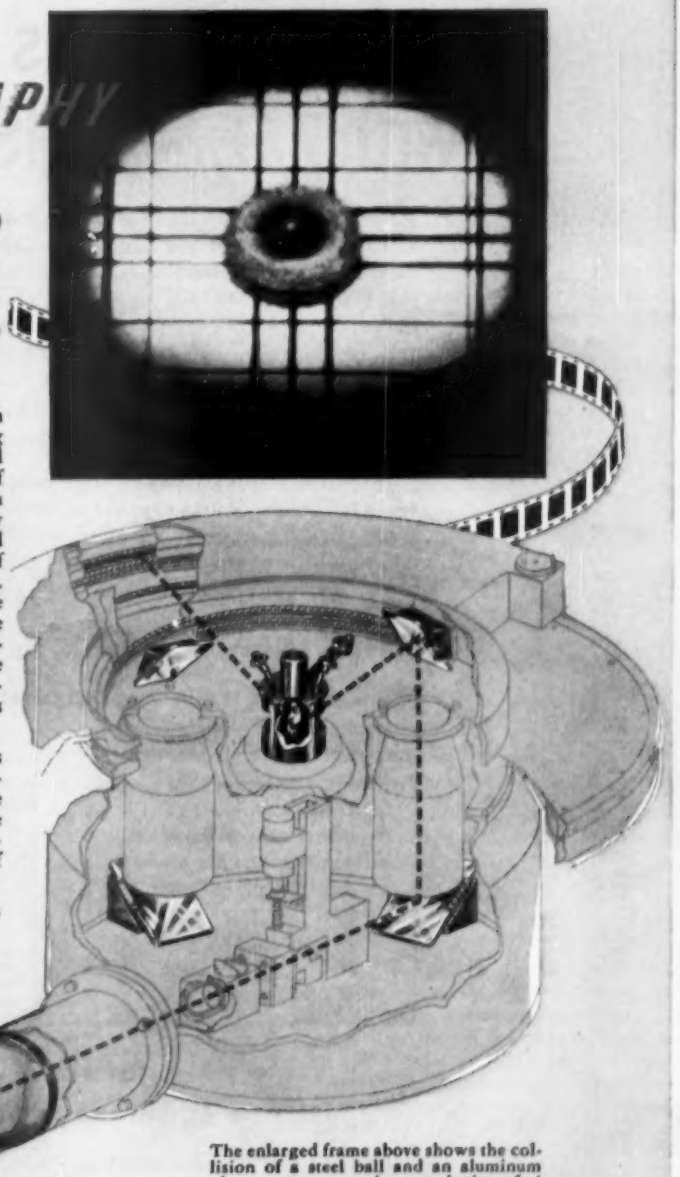
15,000,000 PICTURES/SECOND



Here at Los Alamos, the development of high speed photography has produced framing cameras of unprecedented framing rates and exposure times. These cameras are capable of taking as many as 90 frames at rates as high as 15 *million* frames a second. They employ the technique of sweeping the image, reflected from a rapidly rotating mirror, over a set of correcting lenses onto the recording film. This results in the effective stopping of image motion within the frame. In addition to the creation of new optical components, the construction of these cameras has involved the development of techniques for rotating mirrors of substantial size at speeds as high as 22,000 revolutions *per second.*

Used in a wide variety of research programs as well as in the Laboratory's weapon investigations, instruments such as these typify the excellent resources, in facilities and in the capability for creating wholly new experimental methods, enjoyed by the scientists of Los Alamos.

For an illustrated brochure about Los Alamos write to:

Director of Personnel
Division 1808

The enlarged frame above shows the collision of a steel ball and an aluminum plate at an approximate velocity of 4 millimeters/microsecond, illustrative of studies of interaction of metals at high impact velocity. The cutaway drawing shows some of the features of one of the Laboratory's high speed framing cameras.

los alamos
scientific laboratory
OF THE UNIVERSITY OF CALIFORNIA
LOS ALAMOS, NEW MEXICO

You get millions of feet of nuclear with *Carpenter* Stainless and High

● Carpenter's role in harnessing the atom, first for wartime use and later for national defense and peacetime uses, began during World War II. For nearly 15 years, Carpenter has cooperated with the AEC and prime contractors on many of the nation's top nuclear energy projects, supplying millions of feet of stainless and high alloy tubing and pipe for these installations.

Carpenter's leading role in the nuclear energy field is shown by the following applications: Carpenter supplied stainless tubing and pipe for Hanford Works' first production facility; the first finned tubing used by the AEC at Hanford Works; 100 miles of tubing supplied for the Dana Project; square-shaped tubing to Argonne National Laboratory for a boiling water reactor; and many more.

Significant Carpenter developments include the first successful use of welding to produce stainless and special alloy tubing and pipe . . . the origination of Schedule 5 pipe . . . super corrosion-resistant Stainless No. 20Cb . . . titanium tubing and pipe . . . long lengths of Alloys B and C . . . and stress corrosion-resistant 7Mo Stainless Tubing and Pipe.

More than 30 analyses of tubing and pipe can be and have been supplied for nuclear energy plants

Carpenter experience in producing millions of feet of tubing and pipe for nuclear energy requirements embraces these standard stainless analyses and specialty high alloys:

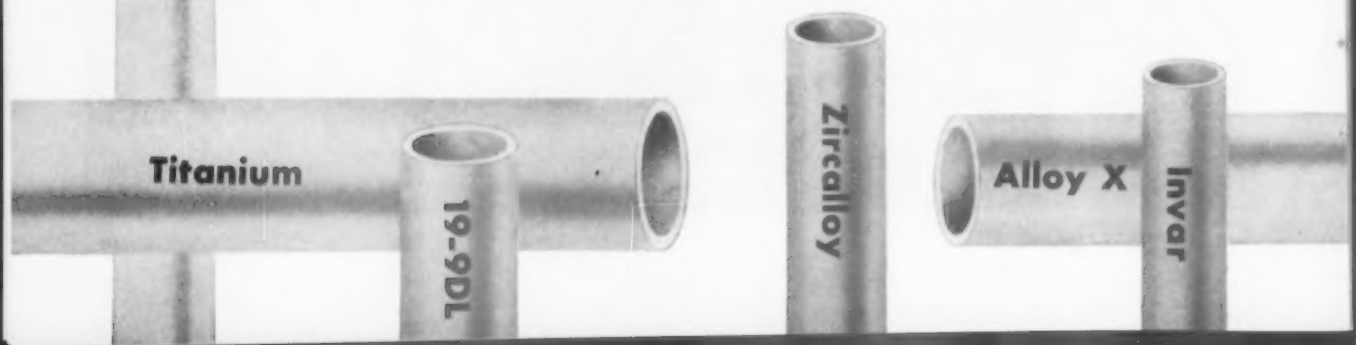
304	310	317L	348
304L	316	318	430
309	316L	321	442
309S	317	347	443
309S-Cb			

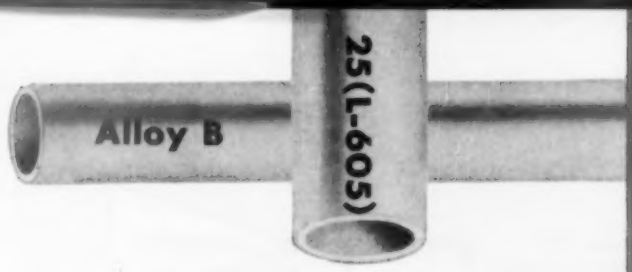
SUPER STAINLESS AND HIGH ALLOY GRADES

Zircalloy II	*Hastelloys:*	Invar "36"
Titanium C-55 or C-70	Alloy B	Invar "42"
Stainless No. 20Cb	Alloy C	HiMu "80"
Carpenter 7Mo (329)	Alloy X	430 Ti
19-9DL	Alloy F	N-155
19-9DX	25 (L-605)	

Wide range of sizes, gauges, lengths, forms and finishes for nuclear requirements

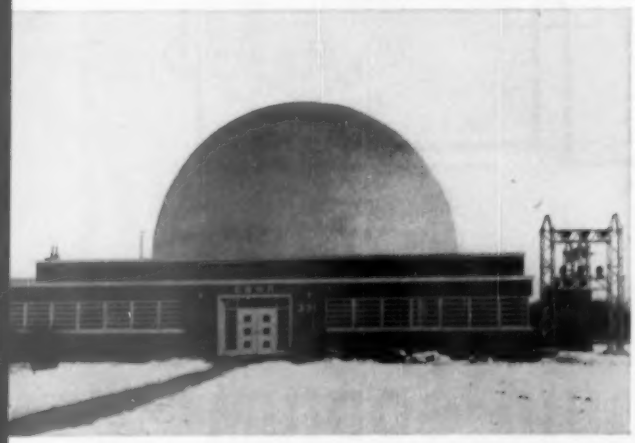
Tubing and pipe sizes available from Carpenter are as follows: *Tubing* from ¼" to 4½" O.D. with 35 BWG (.005") to 9 BWG (.154") walls. *Pipe* of Schedules 5 and 10 in sizes of ⅛" to 4" and Schedule 40 up to 2" IPS. Finishes range from as-welded to high polished. Regular lengths to 45 ft. with longer lengths when specified. Many shapes other than round can be supplied to order.





experience Alloy Tubing and Pipe

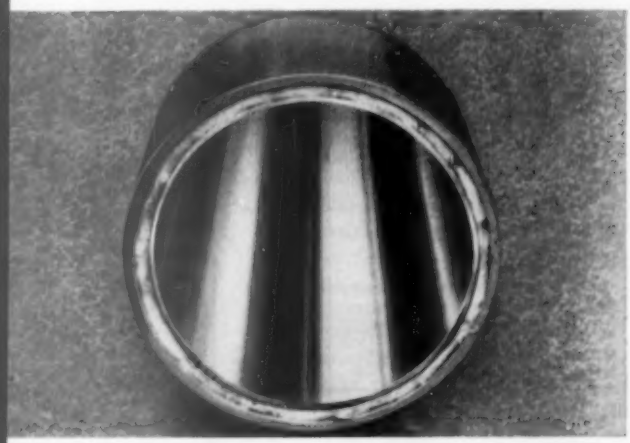
Carpenter Stainless Tubing is being used in many heat exchangers such as this one for nuclear energy applications. Carpenter quality and ability to meet extremely close dimensional tolerances make Carpenter Tubing ideal for use in heat exchangers.



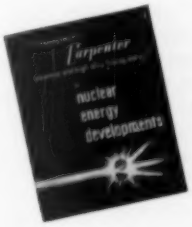
Carpenter has supplied square-shaped tubing for the boiling water reactor in the first nuclear power plant built specifically for electric power generation in the U.S., the 5000-kw experimental plant now in service at Argonne National Laboratory. Illustrated is a front view of the Experimental Boiling Water Reactor. Carpenter also supplied the tubing for the primary heat exchanger for a similar commercial-size plant under construction for a large Mid-West utility.

Photograph courtesy of General Dynamics Corporation

Reprocessing nuclear fuel on the Submarine Intermediate Reactor for the "Seawolf" calls for the use of super corrosion-resistant Carpenter Stainless No. 20Cb in tubing, sheet and plate forms.



Extremely close dimensional tolerances and polish requirements were met by Carpenter in nearly 500 feet of stainless tubing installed in an AEC nuclear power aircraft test facility. Compressed air forces radioactive particles through the tubing, which passes through a 32½ ft. thick concrete wall shielding the reactor. Temperature inside the tubing during operation ranges between 1600° F and 2000° F.



If you are concerned with the design, construction or application of nuclear power equipment requiring tubing and pipe, draw on the broad and extensive atomic experience of Carpenter's technical staff and production facilities. Plan to see our representative soon and send for Bulletin.

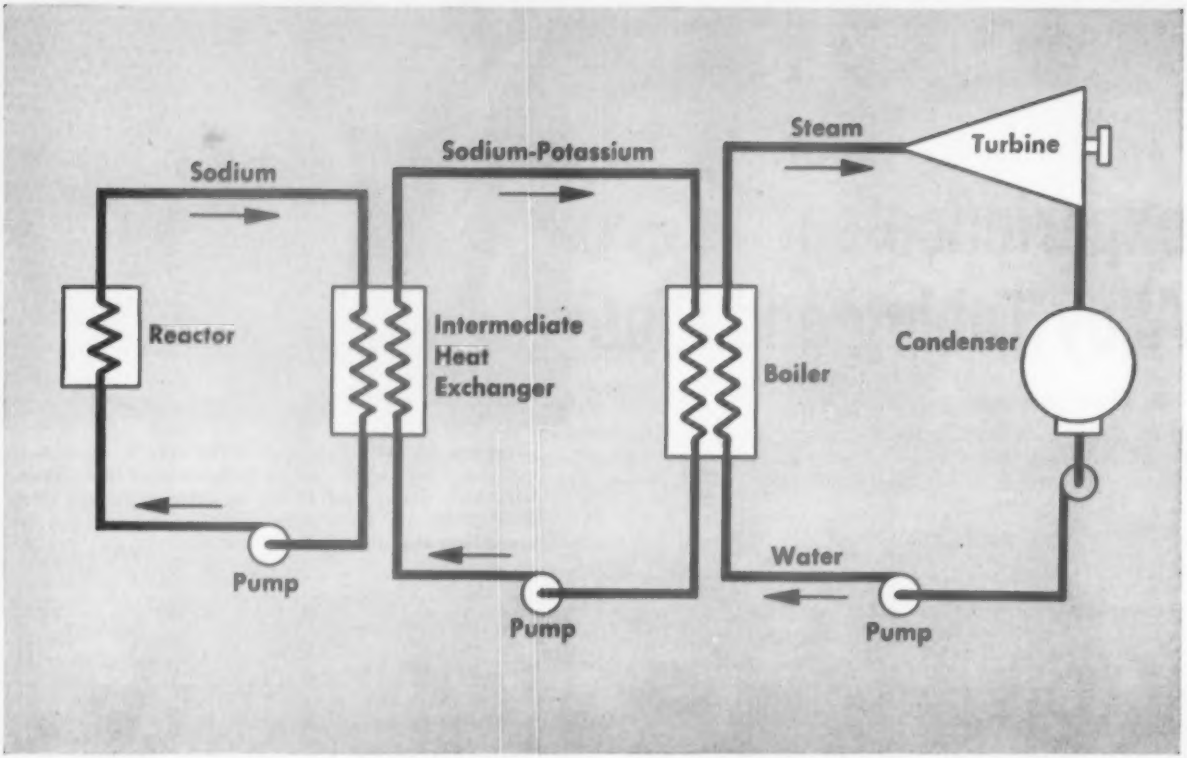
MEMBER

The Carpenter Steel Company
Alloy Tube Division, Union, N. J.

Export Dept. The Carpenter Steel Co., Port Washington, N. Y.—"CARSTEELCO"

Carpenter

Stainless Tubing & Pipe



What makes LIQUID SODIUM a first-rate Reactor Coolant?

At least 6 nuclear power plants now completed, planned, or under construction will use sodium metal in liquid form to transfer heat from reactor to steam generation circuit. Why have the designers of these systems selected sodium for the job?

Because sodium is a liquid from 207° to 1621° F., a sodium-cooled system can operate at high temperatures without expensive pressurization. In this way it can yield steam cycle efficiencies equivalent to those of conventional power plants.

Liquid sodium can be pumped and circulated as easily as water. Further, because liquid sodium has good electrical conductivity, it can be circulated with an electromagnetic pump containing no moving parts. With the low melting point of 207° F., it won't solidify in the heat exchanger system if the reactor is slowed down.

Sodium has high heat conductivity, as well as good radiation stability and high temperature stability. In the absence of oxygen, water, and other impurities, it can be handled without corrosion in iron and steel equipment.

U.S.I. supplies more sodium to outside industry than any other producer of the metal. For complete information on sodium and its handling, send for the free 40-page booklet "HANDLING METALLIC SODIUM ON A PLANT SCALE".

SOME REACTORS WHICH WILL USE SODIUM:

(1) **California**—sodium-graphite reactor, 7,500 kw
(2) **Nebraska**—sodium-graphite reactor, 75,000 kw
(3) **Idaho**—fast breeder reactor, 15,000 kw
(4) **Michigan**—fast breeder reactor, 100,000 kw
(5) **Alaska (Anchorage)**—sodium-deuterium reactor, 10,000 kw
(6) **Seawolf**—sodium-beryllium reactor
(7) **New York**—sodium-beryllium reactor

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Division of National Distillers and Chemical Corporation
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nuclear instruments

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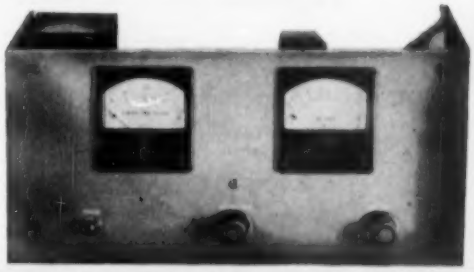
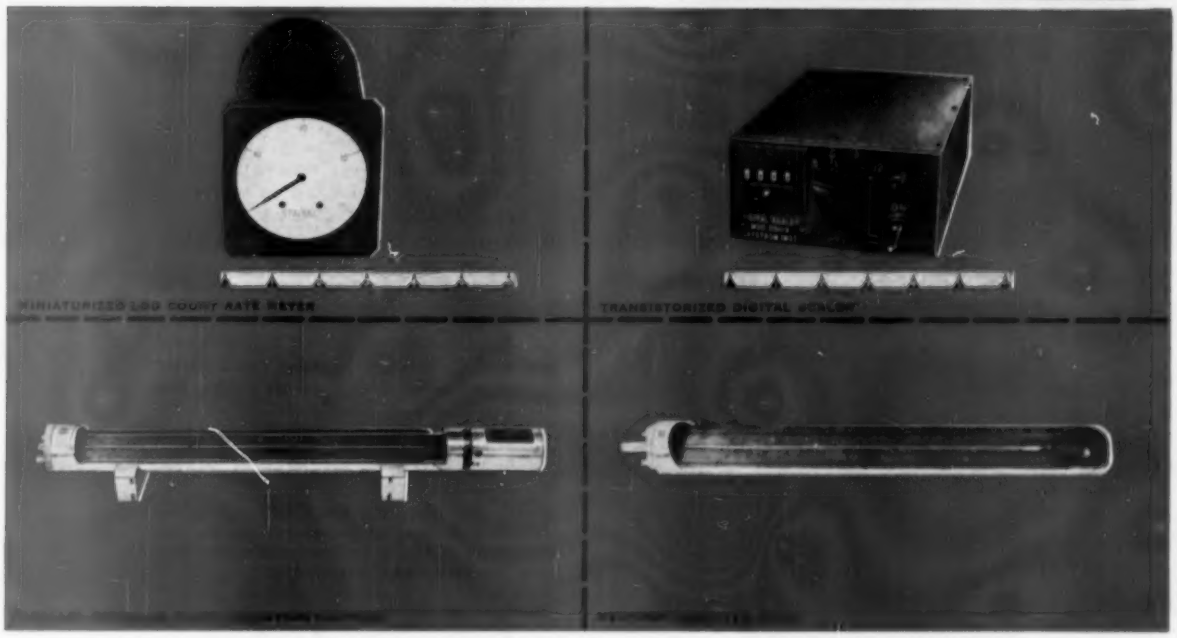
To meet the ever increasing demand for quality instruments of proven design, Daystrom Nuclear has consolidated its engineering and production capabilities . . . Daystrom's scientific and engineering "know-how" in nuclear instrumentation is being utilized to produce an improved line of instrumentation at the expanded facilities in West Caldwell, New Jersey . . . Skilled technicians, using efficient production techniques, are meeting the challenge of the nuclear industry to provide miniaturized and newly packaged instruments of high reliability and advanced design .



LOG COUNT RATE METER WITH PERIOD



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Daystrom can meet your requirements from stock, or design to your specifications, a single component or a complete system . . . quickly and at competitive prices. Write today for Engineers' Reference File Folder containing technical information for reactor control instruments.

ultra-flexible system continuously monitors radioactive airborne particulates

Tracerlab map-1

The MAP-1 Monitor for Airborne Particulate Radioactivity is flexible in design so it can be installed in a permanent station or made mobile. The filter paper movement can be continuous or step. Scintillation or Geiger-Mueller detectors, linear or log ratemeters are used to meet the requirements of any particular problem. It combines in one complete system the five primary requisites for population and personnel protection and legal evidence:

1. ***Efficient Sampling*** The sampler in the MAP-1 system draws a continuous air sample through a moving filter paper by means of a vacuum pump at rates up to 10 cfm. The filter paper traps 98% of all particles 0.3 microns and larger.
2. ***Sensitive Detection*** The detectors in the MAP-1 system count alpha radiation in concentrations as low as 10^{-12} μc/cc; beta gamma radiation as low as 10^{-10} μc/cc. A Geiger-Mueller counter monitors beta and gamma radiations directly on the filter paper. Alpha particles deposited on the filter paper are continuously detected by a scintillation counter. Output of these detectors is fed to ratemeters and recorders, which may be located in a remote control room or mobile cart.
3. ***Automatic Operation*** The MAP-1 system can be operated unattended for periods up to one week.
4. ***Functional Alarms*** The system's key functions are monitored by visible and audible alarms which signal filter paper tear, low flow on air intake, and excessive radiation level.
5. ***Operational Checks*** Sources are provided to check the response of the radiation detector system.

Write for Bulletin RMS-100 for more detailed technical information on the Tracerlab MAP-1 Continuous Monitor System for Airborne Particulates.

Tracerlab

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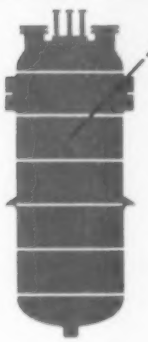
2030 Wright Avenue, Richmond 3, California

Offices in principal cities throughout the world



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"TAILORS OF NUCLEAR STEEL" ®

CAMERA . . . . . ACTION!

The 3⅜" stainless clad steel upper course of this Boiling Water Reactor is completely covered with X-Ray film. The Gamma Ray unit in the center is ready to prove perfection of steel and weld . . . or spot trouble at this stage. This Gamma Ray operation is one essential test, following a Paceco inspection calendar in excess of ASME Code requirements. All plate for the entire reactor shell and heads is completely X-rayed. All welds are fully penetrated. Root passes of all nozzle welds are Magna-Fluxed. And the 17" thick x 9' diameter flanges are ultrasonically tested.

Basic size of this 60 ton reactor vessel was determined by the customer – General Electric Co. Entire design and engineering was supplied by Paceco.

Here are 4 Paceco gained-through-experience skills you may be seeking for design and fabrication of your nuclear equipment:

1. Skill in welding and heat treating of steel and alloys, both light and heavy gauge.
2. Skill in fabricating of corrosion-resistant metals and alloys.
3. Skill in producing heavy equipment to extremely precise mechanical tolerances.
4. Skill in designing and building equipment to resist extreme pressures and intense temperatures.

Some units we have built direct from buyer's plans. In other instances, however, Paceco engineers work with the buyer *at the designing stage*. Experience has proved this consultation may sharply lower initial cost, improve operational efficiency and reduce maintenance. We invite consultation on your requirements.

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On this subject:

HEAT EXCHANGERS FOR NUCLEAR SYSTEMS

. . . Ross speaks with experience

The initiative which Ross has taken in virtually every field of heat exchanger development applies to the nuclear energy field, as well. Ross engineers were early at work.

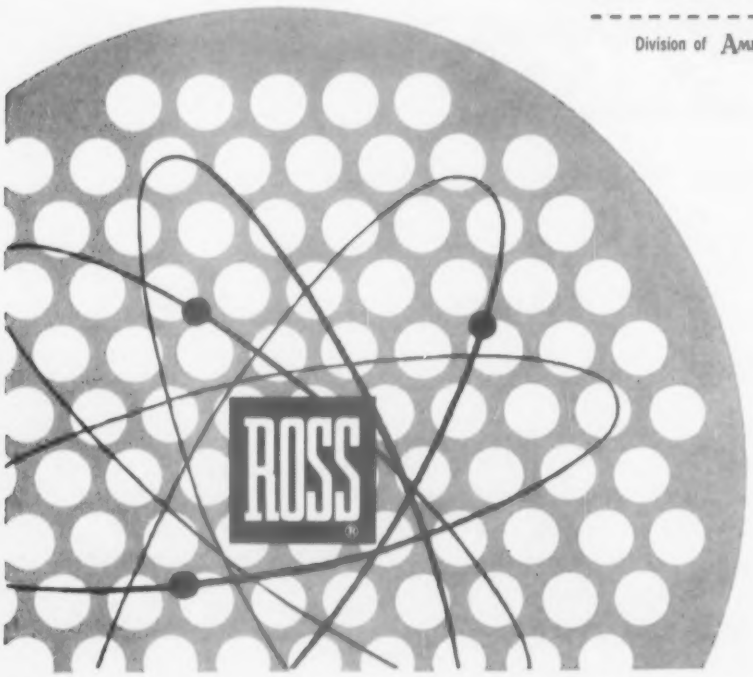
Starting with exchangers designed and fabricated for the Manhattan Project, and carrying on down right through to today, Ross has been steadily building its reputation in nucleonics' circles.

The readiness of Ross to perform engineering development; the possession of the necessary skills to work with specialized materials; the ability accumulated through experience to meet stringent specifications . . . these and others are points which Ross engineers would be pleased to discuss with you. Yes, Ross can speak with experience. Your inquiries are invited.

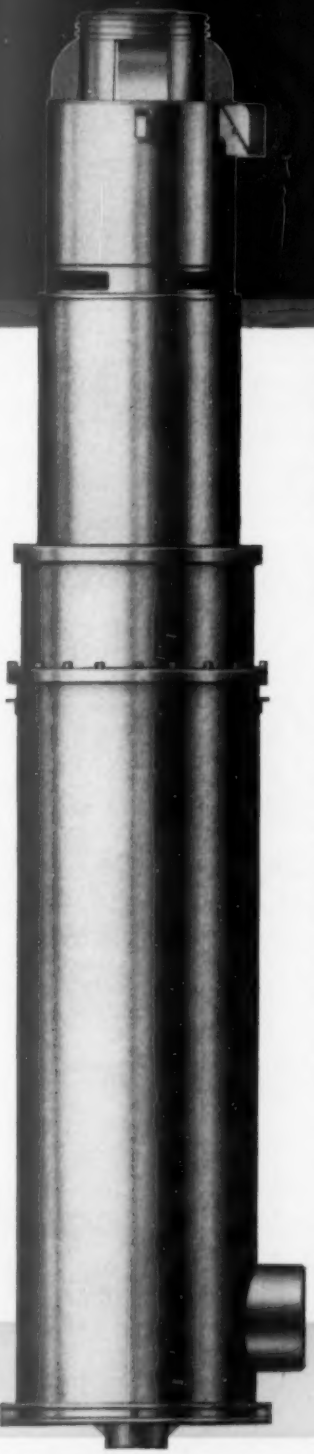
Ross Heat Exchanger Division of American-Standard, Buffalo 5, N. Y.
In Canada: American-Standard Products (Canada) Limited, Toronto 5, Ont.

ROSS HEAT EXCHANGER

Division of AMERICAN-Standard



Another Byron Jackson FIRST in NUCLEAR PUMPS...



BJ Liquid Sodium Pump

WORLD'S LARGEST LIQUID SODIUM PUMP FEATURES MANY DESIGN "FIRSTS"

This new Byron Jackson pump will be used as a primary cooling pump for one of the nation's first full-scale commercial power plants using nuclear fuel.

The pump is approximately 32 feet high with a 6 foot diameter. Rated at 1000 hp, it will pump liquid sodium at a temperature of 1000° F. and deliver 11,800 gpm against a total dynamic head of 310 feet.

Of special interest are Byron Jackson's engineering accomplishments in designing the pump and installation for "packaged" removal. Basically the pump is divided into two sections—(a) suction tank incorporating suction and discharge nozzles which will be permanently welded into the system, and (b) the complete motor and pump assembly which can be pulled as a unit for routine maintenance and inspection with maximum ease, speed and safety. The "packaged" design even provides for shielding (normally separate from the pump assembly) to be integrated within the pump motor support assembly.

This record-making liquid sodium pump with its special design features marks another step in Byron Jackson's continuing leadership in the liquid metal pumping field.

BYRON JACKSON PUMPS
INCORPORATED
A SUBSIDIARY OF BORG-WARNER CORPORATION

P.O. Box 2017A, Terminal Annex • Los Angeles 54, California

Evaluation Study Displays How

NUCLEONICS PENETRATES ALL LEVELS

... Basic Research, Design Engineering, Management

To be important to important industry people a magazine *must* know their specialized activities and work problems.

As part of NUCLEONICS' continuing effort to maintain this contact, detailed questionnaires were recently mailed to every fifth subscriber to the magazine . . . the readers responded with an impressive 25% return.

Note the *diversification of coverage* indicated by their answers to the question . . .

"What Classification Most Closely Fits Your Job Function?"

37% of the respondents said, "Engineering."
35% said, "Scientific."
23% said, "Management."

More specific job areas are revealed by a further breakdown of the two largest categories mentioned.

"Engineering"

19% of the above respondents said, "Nuclear Engineering."
13% said, "Mechanical Engineering."
12% said, "Electronic Engineering."
11% said, "Electrical Engineering."
10% said, "Chemical Engineering."

"Scientific"

26% of the above respondents said they were "Physicists."
11% said, "Chemists."
4% said, "Biologists."
3% said, "Metallurgists."

Findings do *not* equal 100% because only highlights of the question are given.

In the nuclear market, your advertising MUST reach into *each* of these varied levels to exert buying influence. And the study establishes NUCLEONICS ability to do this . . . no matter how diversified their job or background, *key* industry men are grouped together by a *common* tie: a reliance on NUCLEONICS for decision-making information.

Concentrate your efforts in the one publication that best serves their specialized needs: NUCLEONICS. Start your campaign with the August Issue. (Forms close: July 8.)

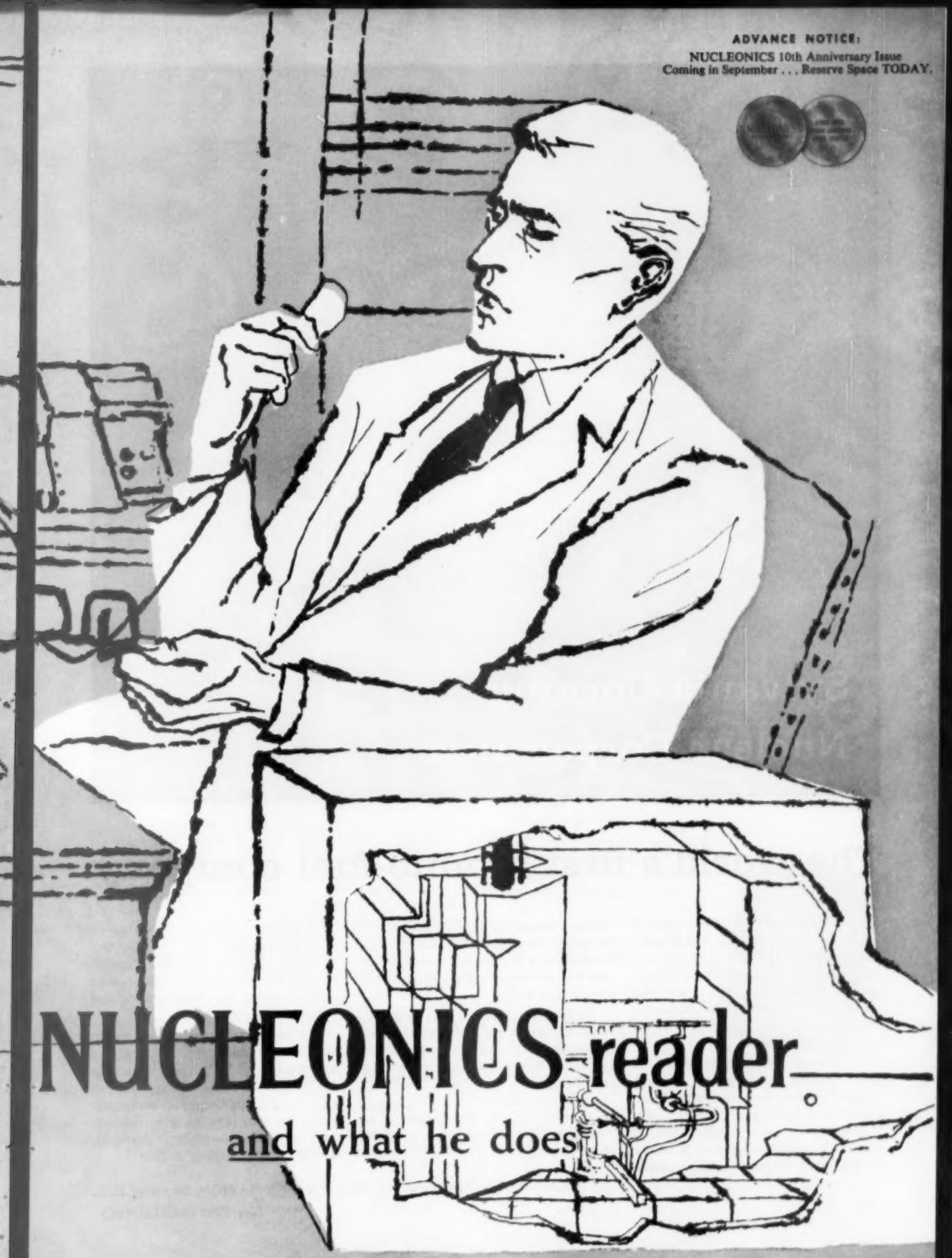
Review the entire findings of Report #81 yourself. Send for a copy of "The NUCLEONICS Subscriber . . . He Tells Us Who He Is."

NUCLEONICS ABC ABP

a McGRAW-HILL PUBLICATION, 330 West 42nd Street, New York 36, N. Y.

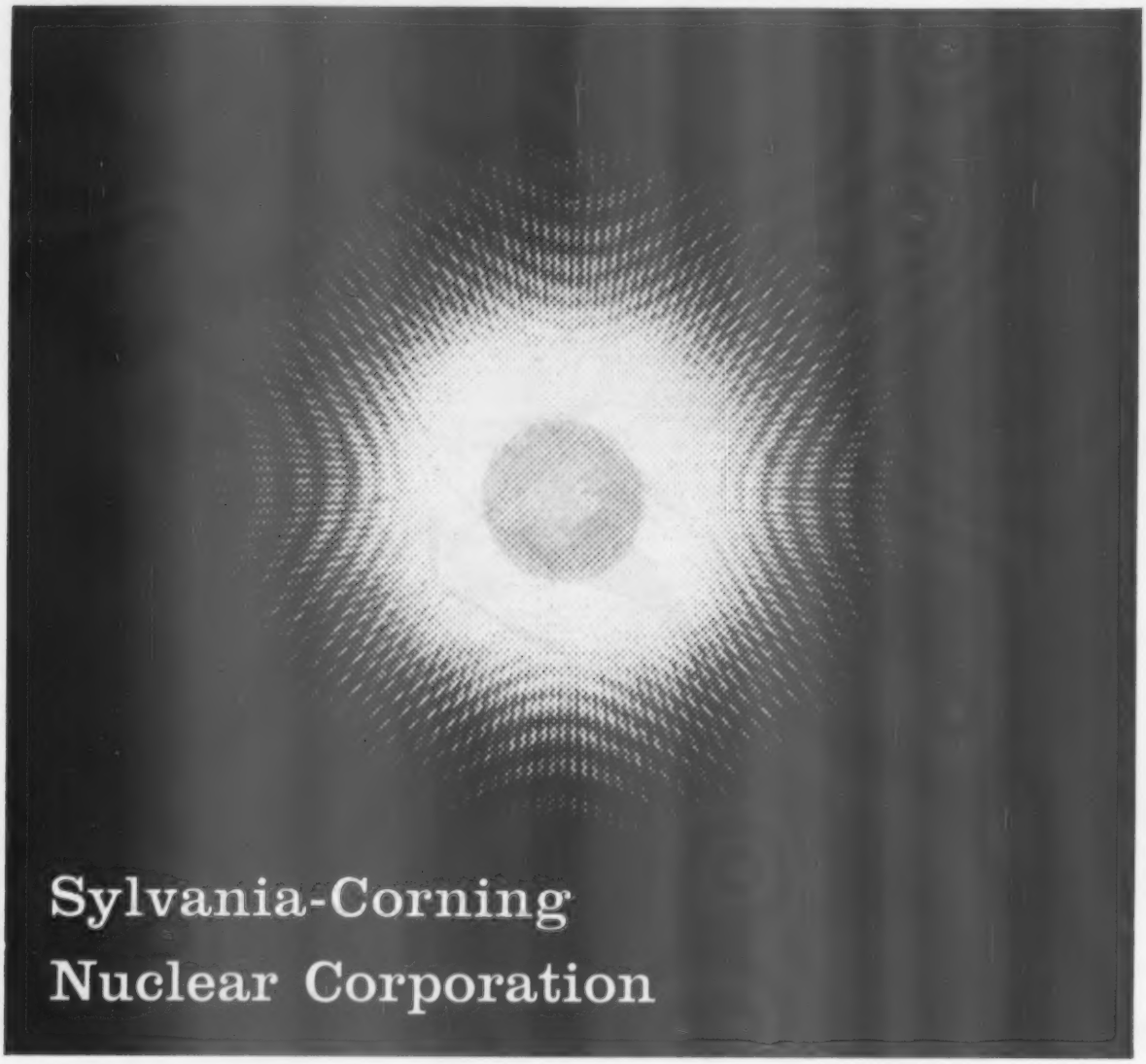
NEW . . . RESEARCH REPORT #81 . . . Defines the

ADVANCE NOTICE:
NUCLEONICS 10th Anniversary Issue
Coming in September . . . Reserve Space TODAY.



NUCLEONICS reader

and what he does



The world's first atomic fuel company

Now that the energy of the atom is being successfully harnessed to commercial uses, the world is rapidly moving ahead to exploit this huge source of power. Several nuclear power plants are already producing electricity. Many more are being built the world over.

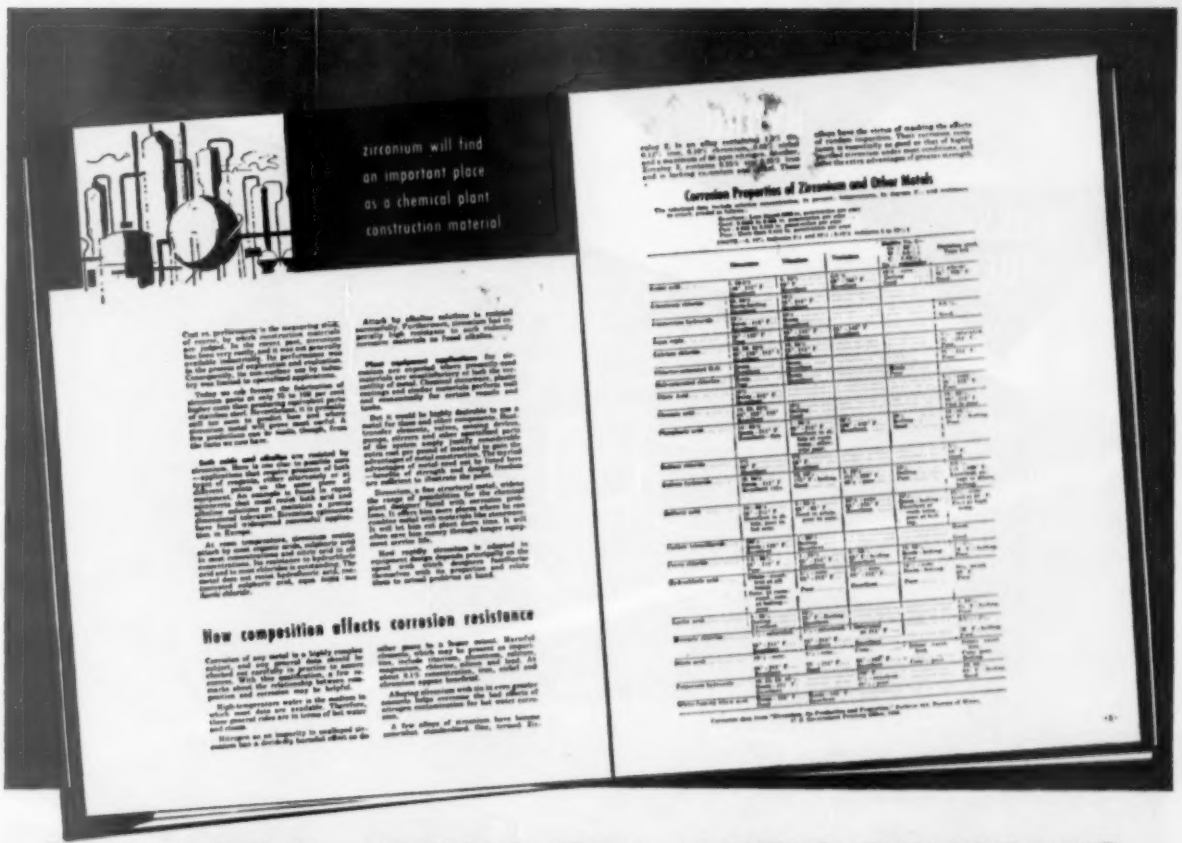
To supply the growing need of these atomic power plants for any type of nuclear fuel, Sylvania Electric Products Inc. and Corning Glass Works have formed the jointly owned Sylvania-Corning Nuclear Corporation. Through specialization in the design, fabrication and processing of nuclear fuels, the Corporation's aim is to provide complete and continuous service to all of the users of nuclear fuels.

Sylvania's Atomic Energy Division brings to the new corporation more than eight years of experience in the design and fabrication of atomic fuels and fuel recovery processes. Corning Glass Works has pioneered in the development of high-temperature ceramic nuclear fuel elements.

The nuclear energy activities of both companies have become the Sylvania-Corning Nuclear Corporation, the only company devoted exclusively to the development and supply of nuclear fuels.

For more complete information on this new organization, write to the address below. The combined scientific, engineering, and technical staff will be glad to help you solve any problems you may have in nuclear fuel application—whether immediate or in the planning stage.

SYLVANIA-CORNING NUCLEAR CORPORATION, Bayside, N. Y.


zirconium will find
an important place
as a chemical plant
construction material
How composition affects corrosion resistance
Corrosion Properties of Zirconium and Other Metals

WHAT'S NEW ON ZIRCONIUM?

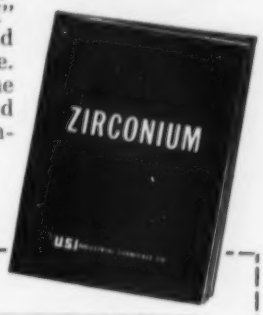
This U.S.I. booklet gives you information you can use . . . Send for a copy!

Today, with zirconium production trending sharply upward, and price downward, interest in this new and unique metal is growing rapidly.

To answer your questions about zirconium and its relative, hafnium, U.S.I. has prepared an informative new booklet — "ZIRCONIUM and HAFNIUM." The booklet gives the important facts on these newly available metals . . . lists important mechanical and physical properties of both . . . describes briefly the new semi-continuous sodium reduction process for production of zirconium and hafnium.

You'll find the outstanding corrosion properties of zirconium compared with other important metals... phase diagrams of twenty binary alloy systems . . . descriptions of several fabrication techniques.

You're sure to find "ZIRCONIUM and HAFNIUM" valuable in discussions and as a preliminary reference. For your copy, forward the coupon today . . . or just send your name on your company's letterhead.



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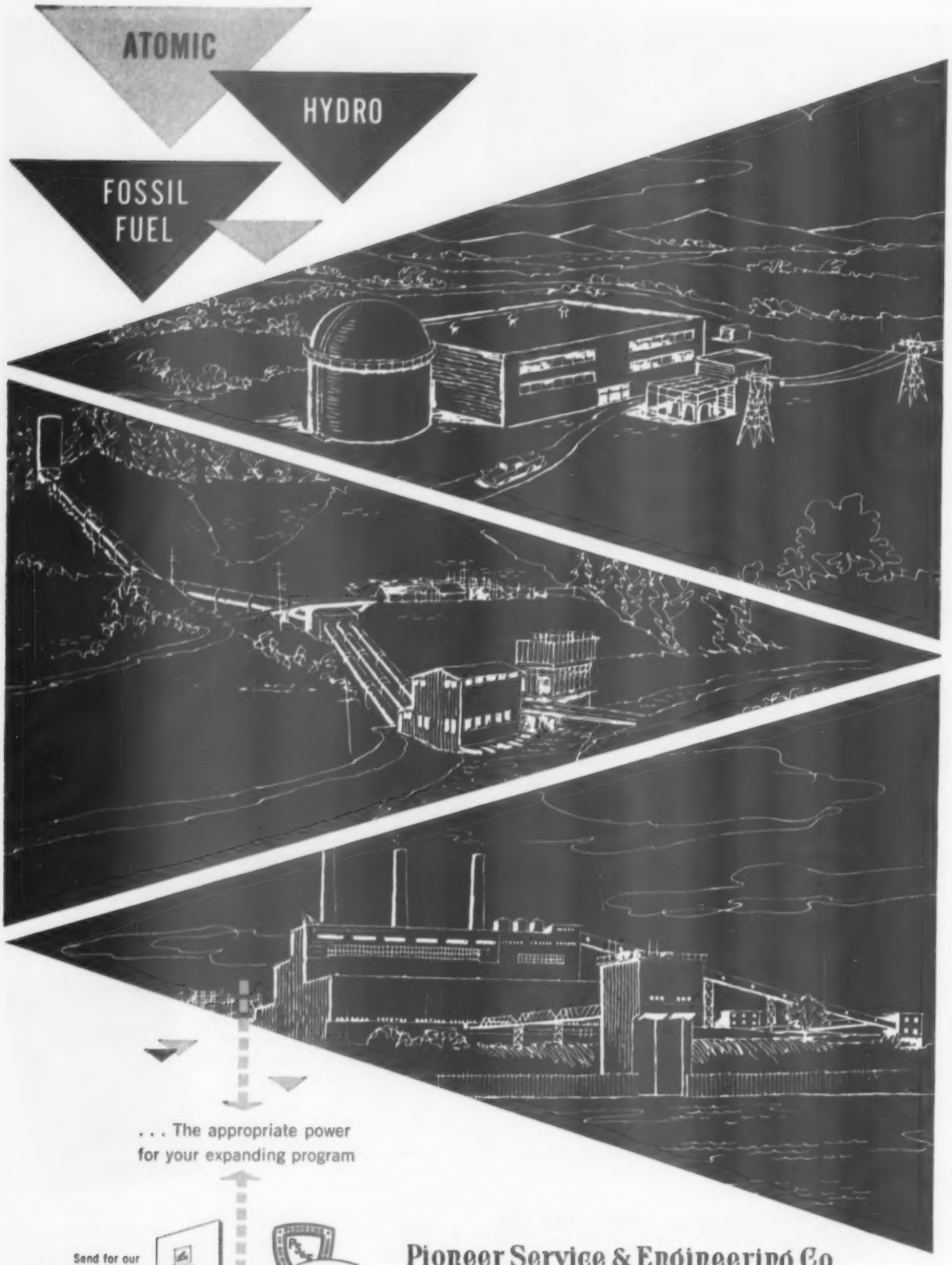

. . . The appropriate power
for your expanding program

Send for our descriptive booklet, "Pioneering New Horizons" of these and other services.

55 YEARS

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What—

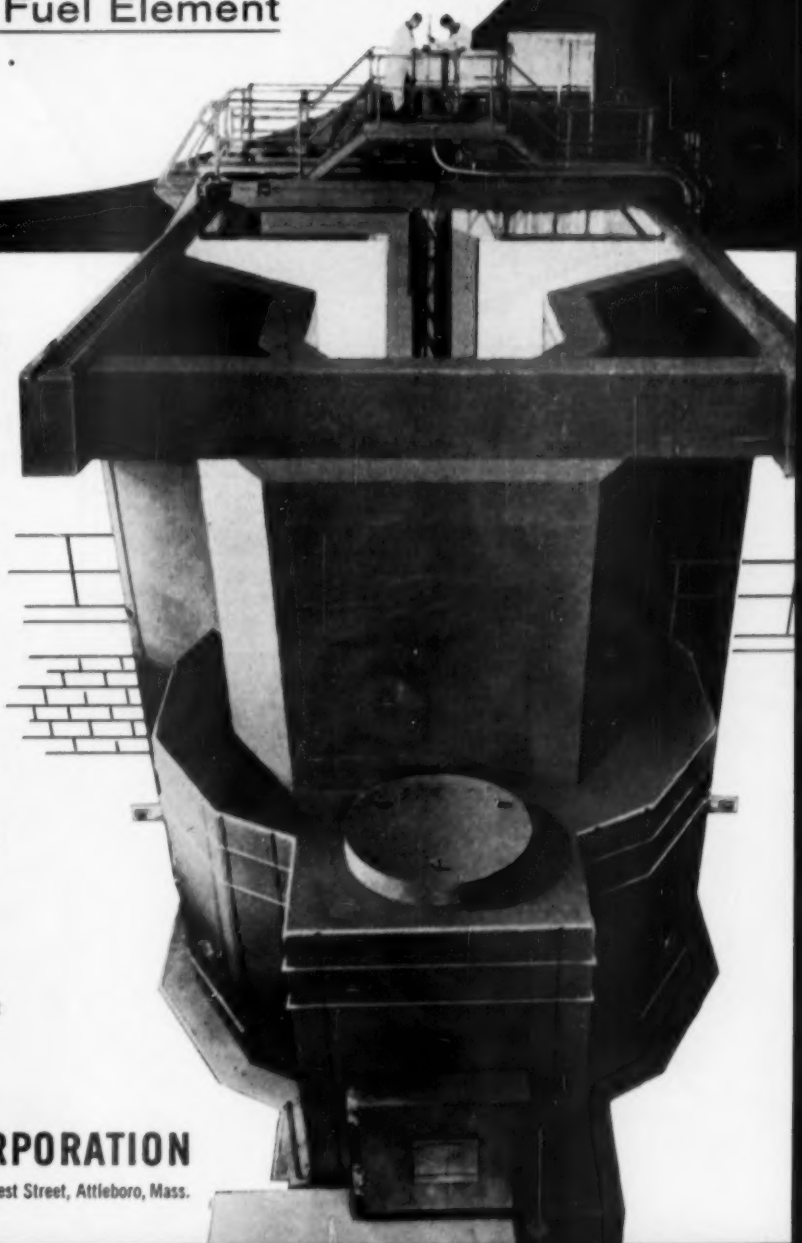
Metals Fabrication

When—

Since 1916, a manufacturer experienced in the handling of new and unusual metals

Who—

Metals & Controls Corporation, Leader in the field of Nuclear Fuel Element Production . . .



Metals & Controls Corporation, with over 40 years of experience in custom type metal fabricating operations, was logically an early entrant into the nuclear fuel industry. From rolling gold plate for the pre-World War I jewelry industry, to supplying industrial metals for jet age aircraft and electronic applications, Metals & Controls has grown in diversification to meet the nation's requirements for the fabrication of new and unusual metals.

It is in this tradition that the Nuclear Products Division of Metals & Controls Corporation maintains its position in the nuclear fuel element field. Modern equipment is constantly being added to existing facilities to make better products while continuing to use basic metallurgical techniques. Scientific people with an interest in progressive metallurgy are needed to carry out programs utilizing this modern equipment.

The aggressive pursuit of new and better ways to process metals into better and simpler fuel element designs is the aim of the rapidly expanding production staff at Metals & Controls Corporation.

If you are interested in joining this forward moving company, write to our Employment Director for the new booklet, "Your Opportunities at Metals & Controls Corporation." We will gladly furnish you with a copy.

Or, if you are a manufacturer with nuclear fuel element needs, your inquiries on fuel elements for research or power reactors will be given our closest attention. Contact our Manager of Contract Administration. He will be pleased to consult with you on your requirements.

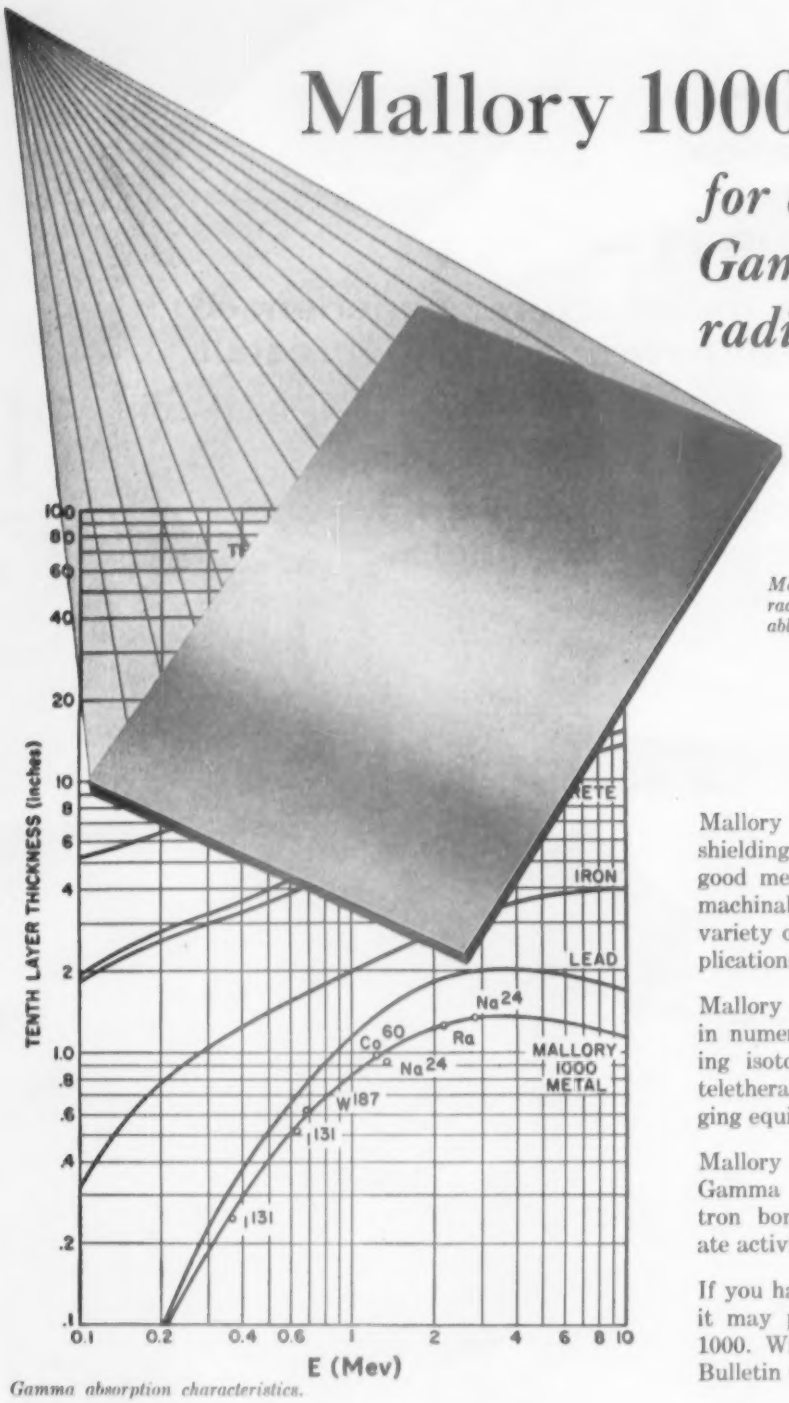
METALS & CONTROLS CORPORATION

Nuclear Products Division

1306 Forest Street, Attleboro, Mass.

Mallory 1000 Metal

for effective Gamma radiation shielding



Gamma absorption characteristics.

Mallory 1000 "slab" for radiation shielding available in sizes up to 6" x 15".

Mallory 1000 Metal is ideal for radiation shielding applications. It has high density, good mechanical strength, and excellent machinability . . . is available in a wide variety of shapes and sizes for many applications in this field.

Mallory 1000 Metal is already being used in numerous applications such as shielding isotope shipping containers, cobalt teletherapy machines, and oil well logging equipment.

Mallory 1000 Metal effectively absorbs Gamma radiation . . . low intensity neutron bombardment causes only moderate activity.

If you have radiation shielding problems, it may pay you to investigate Mallory 1000. Write for your copy of Technical Bulletin 6-7C.

Expect more . . . get more from

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When piping changes direction, protect process equipment with

ZALLEA SELF EQUALIZING UNIVERSAL EXPANSION JOINTS

Zallea Self Equalizing Universal Expansion Joints absorb any movement in any direction . . . axially, laterally, angularly, or any combination of these . . . making them ideal for protecting process vessels and rotating equipment from thermal expansion of connected piping.

When there is a leg or 90° offset to a long run of pipe, one Zallea Universal Expansion Joint will frequently take care of the entire system.

Under conditions of low load requirement for anchors or equipment . . . or when compressive loading in pipe walls must be kept to a minimum . . . Zallea Universal Expansion Joints are the most economical answer to the problem. They absorb the internal pressure thrust. The piping system need only carry the relatively small deflection loads.

Many types of Zallea Tied Expansion Joints are designed especially to absorb lateral deflection or rotation when there is pressure in the line. Universal, Modified Universal, Hinged, Gimbal, Pressure Balanced and Universal Pressure Balanced Expansion Joints give you a choice of the type best suited to your particular application.

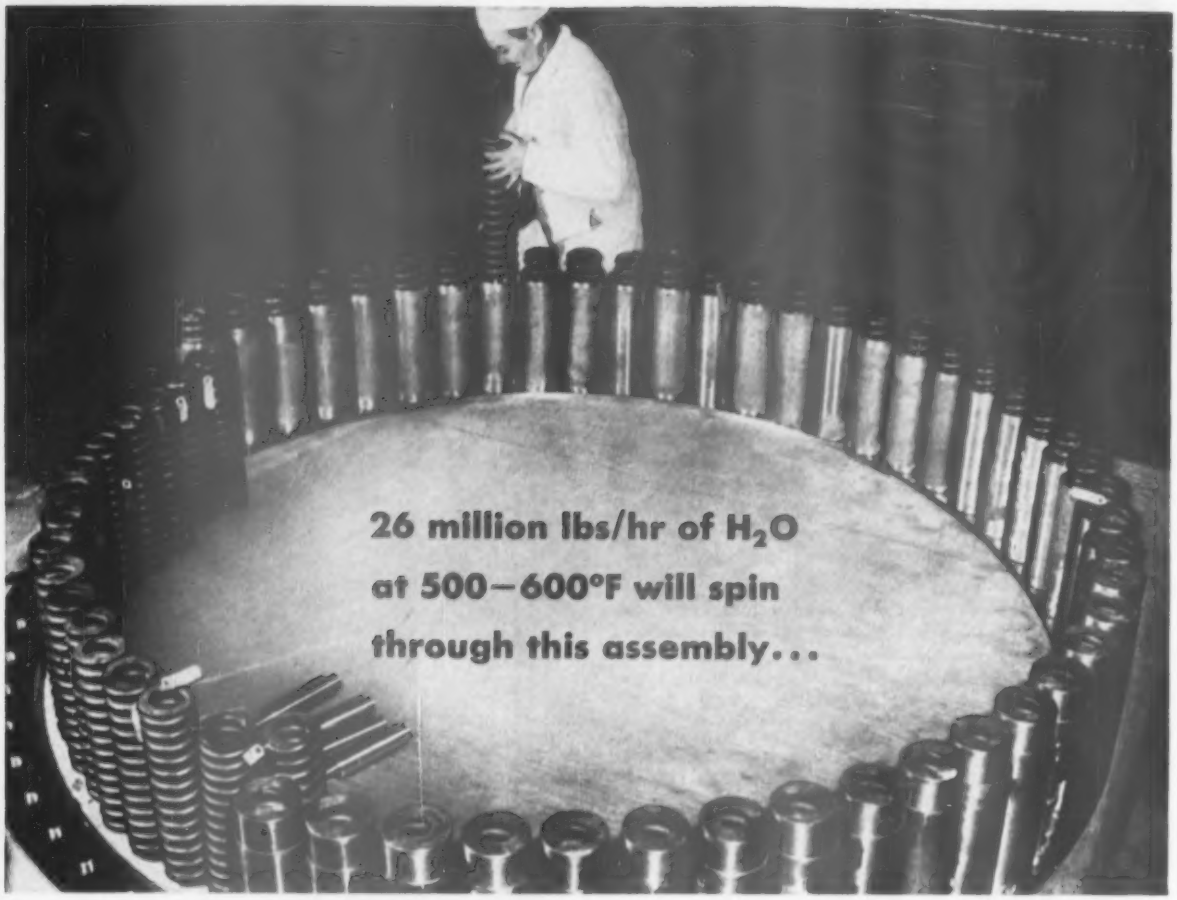
Get the complete story of Zallea Expansion Joints in our new 72-page manual. Write, on your company letterhead, for your copy of Catalog 56. Zallea Brothers, 879 Locust Street, Wilmington 99, Delaware.

Zallea
expansion joints

Zallea Brothers • Wilmington 99, Delaware

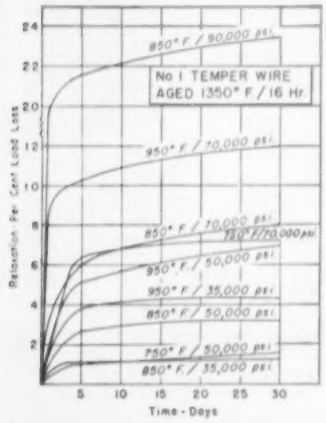
World's largest manufacturer of expansion joints

24" dia. Zallea Universal Self Equalizing Expansion Joints in service lines of the engine test facility of a West Coast aircraft manufacturer.



Westinghouse makes sure core mount springs stand up in Shippingport reactor...makes them of Inconel "X"

Superheated high-purity water makes many ordinary spring materials relax quickly. It may cause them to fail by stress-corrosion cracking or embrittlement.


No. 1 TEMPER WIRE
AGED 1350° F / 16 Hr.
850° F / 90,000 psi
950° F / 70,000 psi
850° F / 70,000 psi
750° F / 70,000 psi
950° F / 50,000 psi
950° F / 35,000 psi
850° F / 50,000 psi
750° F / 50,000 psi
850° F / 35,000 psi
Relaxation Per Cent Load Loss
Time - Days

Relaxation of Inconel "X" springs made from No. 1 temper wire.

An outstanding exception is Inconel "X"* age-hardenable nickel-chromium alloy. Springs of this alloy resist both corrosion and heat . . . up to 1100°F in certain applications.

Here are some relaxation curves, for example, which show what stresses Inconel "X" alloy is capable of handling at various temperature levels.

The curves speak for Inconel "X" alloy. When you want springs that resist relaxation at high temperatures this is the alloy to use . . . just as Westinghouse has done in spring mounting the core of the Shippingport reactor.

Inconel "X" gives you valuable bonus properties too

In addition to excellent spring properties at high temperatures, Inconel "X" alloy offers you (1) excellent corrosion resistance (2) outstanding hardness and wear resistance (3) high fatigue strength (4) high tensile strength (5) low thermal expansion (6) resistance to stress-corrosion cracking.

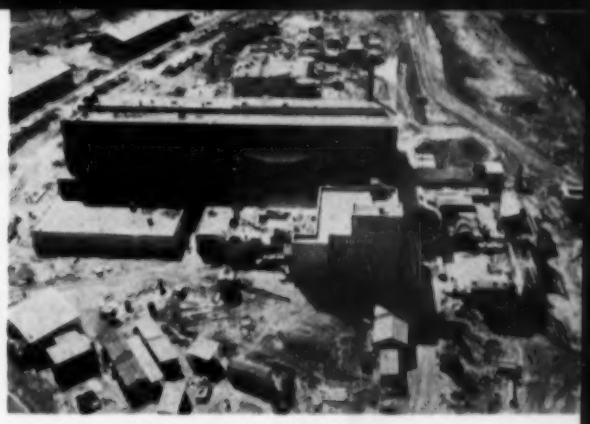
These are just the properties that are so often needed in high-pressure, high-temperature nuclear equipment. You may know Inconel "X" and other Inco high-temperature alloys very well indeed. But if you don't, Inco's 79-page "Inconel 'X' " and 35-page Bulletin T-35 "Inco Nickel Alloy Helical Springs" are two helpful booklets. Write for copies. *Registered trademark

The International Nickel Company, Inc.
67 Wall Street New York 5, N. Y.

INCO TRADE MARK **Nickel Alloys**

Inconel "X" . . . *for durable spring properties at high temperatures*

CALDER HALL, first in a chain of British gas-cooled natural-uranium power reactors



PWR, 60-Mw (electric) pressurized-water reactor at Shippingport, Pa.

World Weighs Nuclear Alternatives

Calder or PWR?

The American pressurized-water reactor and the British Calder Hall gas-cooled reactor have become symbols of the friendly rivalry between these two countries in the field of international atomic power. The British feel that Calder has given them the initial lead in the use of nuclear power and expect to retain that advantage for at least a decade by further improvements. On the other hand, the Belgian government announced purchase of a 1-Mw pressurized-water reactor from a U. S. firm, while an Italian concern is purchasing a power reactor virtually identical with the pressurized-water reactor being designed for the Yankee group at Rowe, Massachusetts.

A technico-economic comparison of these reactor types with emphasis on the situations in the U. S. and Great Britain is presented on page 72 of this issue.

Enriched or Natural Uranium?

Underlying the choice of a pressurized-water or gas-cooled natural-uranium reactor is the broader choice between enriched- and natural-uranium systems generally. This question was considered, with emphasis on solid fuel systems, at a round-table discussion sponsored by the National Industrial Conference Board. Some measure of NUCLEONICS' interest in this subject is indicated by the fact that the original suggestion for such a discussion originated with us. A report on that debate is presented on page 68.

It is important to recognize that these comparisons are limited to certain reactor types and largely ignore the political considerations that can be the real determining factor in some cases. Nevertheless, the complex technical issues must be faced by reactor designers the world over; we hope this symposium in print will help their deliberations.

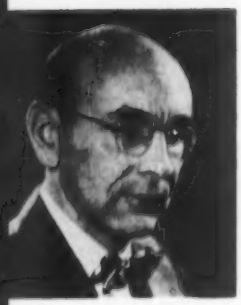
These Five Experts Discuss . . .

JOHN R. MENKE, President, Nuclear Development Corp. of America. NDA is building a 20-Mw natural-uranium-fueled reactor for Alaska's Chugach Electric Association

Enriched vs Natural U

Merits of slightly enriched vs natural-uranium reactors as aired at this debate support the view that natural fuel can hold its own in the atomic power race

W. B. LEWIS, Vice President, Research and Development, Atomic Energy of Canada Ltd. AECL is pursuing several heavy-water-natural-uranium designs



JAMES A. LANE, Director, Reactor Experimental Engineering Division, Oak Ridge National Laboratory, moderated the discussion

CHAUNCEY STARR, Manager, Atomics International Division, North American Aviation Inc., Al's enriched-U reactor projects include the sodium-graphite and organic moderated designs

WILLIAM E. SHOUPP, Technical Director, Atomic Power Department, Westinghouse Electric Corp. APD is developing enriched-U types

THE REACTOR DESIGNS that are being considered for power production in the world today divide into two groups: those that use natural uranium as a fuel and those that require enriched fuel. In both instances, although more radical designs are receiving some attention, the major interest so far has been centered on solid-fuel systems. The technical and economic aspects of the two rival concepts have been hotly debated, most recently at a National Industrial Conference Board meeting in Philadelphia.

The present limited experience with power reactors made definitive conclusions unattainable but it was striking that enriched-uranium systems, so favored in the U. S., were no shoo-in. Conversely, natural-uranium reactors looked better than they are generally given credit for.

J. R. Menke (NDA) summed up this feeling:

"There is no argument that certain engineering advantages, both economic and physical, can flow from the capability of enriching uranium. But the truly surprising thing, the wonderful coincidence, is that one can build such very good reactors with natural uranium. There are even certain parameters which seem to be optimized very near to natural uranium."

An outright prediction today that enriched-uranium reactors will eventually be significantly superior to natural-uranium reactors would seem to depend on a long string of "ifs." Thus the fuel costs of enriched, low-neutron-economy designs can be made to equal the fuel costs of the natural-uranium, high-neutron-economy types *if* fuel-element burnup can be increased by an order of magnitude over present-day figures. Assuming this is accomplished, then enriched reactors will have an advantage *if* their capital investment costs can be made smaller. However, the investment in the reactor itself accounts for only a fraction of such costs. Thus large over-all power-cost reductions can be had only if cuts in investments for conventional components can also be made. Increasing thermal efficiency could be one way to do this *if* the necessary high-pressure-water or liquid-metal system turns out to be not too expensive and *if* natural-fuel reactors do not also take advantage of this approach to the same extent.

For the working of these miracles the enriched reactor is relying mostly on the greater possibility for technical development afforded by its more flexible design requirements. It is always dangerous to make forecasts about what design ingenuity will *not* be able to do. However, it appears that of the design concepts conceived so far none makes an economic advantage for the enriched-uranium reactor a foregone conclusion.

Bases of Comparison

James Lane, chairman of the NICB session, pinpointed the issue by observing that there are really two separate problems to consider in discussing the relative advantages of slightly enriched and natural fuel.

First, for any one reactor system there is a fuel enrichment that gives

lowest total power costs. This optimum enrichment represents a balance between the increased burnup that comes with increased U^{235} enrichment and the higher price paid for the more highly enriched uranium. In this fashion, the optimum enrichment and the penalty for deviating from it should be established for each proposed design.

Second, the reactor types that can be fueled with natural uranium must be compared with the types that require enriched fuels. (Actually the designs usually call for only slightly enriched fuels having up to 4% U^{235} concentration.) Thus we are concerned with the pros and cons of D_2O-moderated or gas-cooled graphite reactors versus the relatively numerous enriched reactor types—pressurized light-water, sodium-graphite, boiling water, etc.

Note that the user of natural uranium is limited to a few reactor types. From these possibilities he must select a design that can compete economically with the reactor types that use enriched fuel. In addition he has to consider the possibility that his chosen design, although it can burn natural fuel, might itself do much better on enriched fuel.

Enriched vs Natural U

With regard to the second question W. E. Shoupp (Westinghouse, APD) presented calculations of fuel costs as a function of enrichment for an Al-D_2O-U system. This system is typical of the kind of reactor that can burn natural U—i.e. a high-neutron-economy type in which burnup is probably limited by reactivity loss.

Shoupp's results are given in Fig. 1. He explained his calculations as follows: "The items that enter into fueling costs are fabrication of the fuel elements, value of new uranium, credit for old uranium, and D_2O charge. The fabrication costs will include the cladding material, the UF_4 to UO_2 conversion and its variation with enrichment, and pelletization of the fuel as well as the actual labor and overhead costs for fabrication. The costs for fuel are all based on uranium as UF_6. Interest is charged on the fuel at 4%. One core is assumed to be in operation while another is in fabrication and reserve. When enrichment fuel is used, an $8/lb UF_6-UO_2 conversion differential is incorporated over that required in the natural-uranium case. The D_2O value is taken at $28/lb and 2 lb of D_2O are assumed to be required for each kilowatt of net electric output. A 14% interest charge on the money required to buy the D_2O is assumed."

Figure 1 shows that for this reactor the use of natural uranium actually results in a slight saving in fuel costs over enriched fuel. For our purposes the essential point is that for this reactor the differences in fuel cost for natural and enriched fuels appear to be negligible (0.2–0.3 mill/kwh).

This seems to be generally true for all natural-uranium-burning reactors—in such high-neutron-economy systems the cost advantage of higher burnup and the cost disadvantage of more expensive fuel roughly cancel.

Comparing Reactor Types

It is much more difficult to make cost comparisons for different reactor types than for one reactor type as a function of enrichment. Such comparisons must refer to the total power costs, including capital investment and operating costs as well as fuel costs. Of these, the major contributions are fuel and capital investment. We will now examine these two factors in turn in spite of the fact that there is little concrete cost experience to draw upon.

Fuel costs. Concerning fuel costs Chauncey Starr (Atomics International) lists these points for and against the enriched reactor:

Advantages—

- Smaller fuel inventory (kg of U)
- Longer fuel burnup than natural uranium if reactivity is limiting
- Possibility of longer fuel burnup when radiation damage is the limiting factor because of the greater choice of fuel alloys

Disadvantages—

- The fuel is more expensive per gram and thus more costly to procure and handle
- Spent fuel cannot be discarded without serious economic penality.

In discussing these points Starr had this to say:

"With regard to fuel inventory charges whether or not a lower quantity of more costly fuel in the reactor leads to lower total costs can only be determined for individual cases. The quantity and value of fuel held up outside the reactor in fabrication and reprocessing can be a more important quantity in inventory costs. For reactors using expensive fuel it is essential that the holdup outside the core be as small as possible. This requires rapid processing and reduction of shipping time."

With respect to reprocessing costs Starr had this to say:

"Suggestions are sometimes made to discard spent fuel elements after irradiation in a reactor in order to avoid the costs of a reprocessing step. For enriched fuel, discarding spent elements is unthinkable, since the remaining U^{235} is far too valuable. The real question is, 'Can *any* fuels be discarded after irradiation?', I think not."

Starr examined the short-range* economic effect of throwing away spent natural-U fuel versus reprocessing it for a natural-uranium gas-cooled reactor. This system should certainly be conducive to the throwaway scheme if any is.

Figure 2 shows the cost per year of throwing away the fuel or reprocessing. The curves show that at any reprocessing cost under $30/kg it is desirable to reprocess rather than throw away. Indications are that reprocessing costs are actually less than this—about $20/kg.

Figure 3 shows similar curves, but with plutonium buy-back price as the parameter and reprocessing costs as the constant (at $20/kg). If plutonium is worth only $6/gm, it would not be worth recovering at this reprocessing cost.

Thus, any combination of low plutonium value and high reprocessing cost favors discarding the spent fuel; higher plutonium value and low reprocessing cost favors reprocessing. Present figures of $12/gm Pu and $20 to reprocess 1 kg U definitely favor reprocessing.

The curves for the throwaway cost in Fig. 2 and 3 do not include any shipping or storage charges. Since any realistic appraisal of the situation indicates that such costs are not negligible, it is apparent that even higher reproc-

* The emphasis on short range is necessary since it is clearly uneconomic to have a long-range atomic energy economy that would throw away fuel after consuming only about 1% of its fissionable content. In addition, it does not seem good business sense to place into permanent storage so much valuable material and mine its equivalent. (See W. H. Zinn, Atomic Industrial Forum Report, July 1954, p. 37).

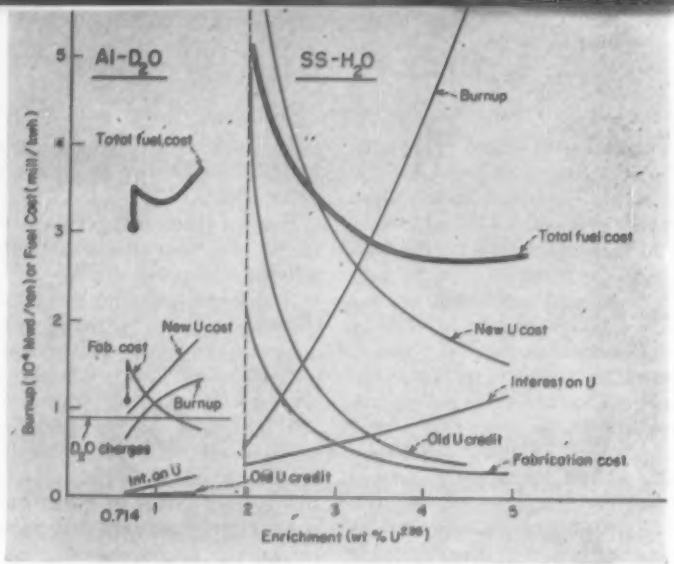


FIG. 1. Total fuel costs as function of fuel enrichment for two different reactor types. Al-D_2O reactor is typical of high-neutron-economy type capable of burning natural uranium. Stainless-steel-H_2O reactor is typical of low-neutron-economy type requiring enriched fuel. Reactor power is 125 Mw in each case. Burnup is assumed to be limited only by reactivity loss

essing charges could be tolerated than indicated in the figures.

As the quantity of reactor fuel handled grows, reprocessing costs should decrease. So again we lean away from the throwaway approach. Thus, we find it very likely that all reactors—natural or enriched—must have the spent fuel reprocessed. This conclusion minimizes one of the alleged failings of the enriched reactor.

Burnup effects. W. E. Shoupp (Westinghouse, APD) provided information on the question of how much influence longer burnup has on fuel costs. Using the assumptions used previously in his Al-D_2O–natural-uranium example, he derived for a stainless-H_2O-U reactor the total fuel cost as a function of enrichment assuming that the burnup achieved at each enrichment is limited by the reactivity available. This system should be representive of the kind of reactor that requires slightly enriched fuel. The results of the calculation are also given in Fig. 1. Unlike the natural-uranium fueled reactor, this reactor type shows substantial reductions in fuel costs thanks to the greater burnup made possible by enrichment. However, the burnups needed to make the fuel costs competitive with unenriched fuel are quite high by today's standards.

With respect to the fuel-cost situation Shoupp pointed out: "For long burnups the costs for the stainless-steel-H_2O reactor can also be 3–4 mills/kwh. In fact, for very long burnups, the fueling costs can actually be lower than in the case of high-neutron-economy reactors. In this case, we are not limited by reactivity but we may be limited by radiation damage and corrosion." . . .

. . . "No one will really know the limitations until reactor cores have actually been run to such burnup levels. It is not possible to justify this point by tests on a few fuel rods or fuel elements."

"At the moment, however, we may conclude that with sufficiently improved solid fuel elements fueling costs of 3–4 mills/kwh are realizable in the future for both Al-D_2O and stainless-steel-H_2O reactor types."

Investment Cost

It was generally agreed that the investment required to build an enriched reactor would be somewhat less than for a natural-uranium reactor of the same power. The reasons are that the reactor itself can be smaller and expensive materials like D_2O and graphite can be avoided.

However Starr pointed out that a large part of the investment in a reactor power plant is in the heat-extraction and turbogenerator systems and in auxiliary apparatus rather than in the reactor. This part of the investment depends only on the total power and not on the reactor type. Consequently the capital investment cost and therefore the total power cost are not usually very sensitive to differences in the cost of building the reactor itself for the various reactor concepts.

There could be an exception to this if the two reactor types have different thermal efficiencies, Menke observed. In such a situation the reactor type with the higher potential thermal efficiency can, by sacrificing a part of its efficiency reduce the size and therefore investment cost of its heat-removal system. Enriched reactors are in a better position to utilize this advantage of high thermal efficiency since they are more easily designed for high-temperature performance. However it is not clear that the cost of the extra design features required for high-temperature operation (high-pressure or liquid-metal systems) will allow a net investment savings.

Apparently, since the reactor part already represents only a modest fraction of the total investment, the rest of which is fairly inflexible, the economics that the enriched reactor can hope to effect through reducing investment costs are rather slim.

Development Potential

Both Starr and Shoupp expressed the feeling that, after all is said and done, the greatest advantage of the enriched reactor approach is that it allows more room for technical developments that can reduce fuel and investment costs.

Starr had this to say: "To the reactor designer, the selection of natural uranium as a fuel severely limits his remaining choices as to moderator, coolant cladding and structure. The inherently low reactivity of the natural-uranium system allows little or no margin for unproductive loss of neutrons. Practically, there are only two choices of moderator—deuterium or graphite. There are only a few choices of low-loss cladding and structural materials such as magnesium, aluminum and zirconium."

"That the designer of an enriched-fuel reactor is not limited is obvious from the descriptions of reactors included in the AEC's five-year civilian power reactor program, and the power demonstration programs. Only a few of the recent proposals contemplate using natural uranium, while the majority use enriched fuel. The very number of enriched

Estimated Power Costs for Natural-Uranium Reactor*

Item	*Rating or quantity*	*Capital investment (10^6)*	*Annual interest charge (%)*	*Operating cost (mill/kwh)*
Turboalternators, transformers and switchgear	200,000 kw	16	8.5	0.97
Reactor structure, auxiliaries and building	800 Mw (thermal)	30	8.5	1.82
Heavy water	3.2×10^5 lb @ $28/lb	9	6	0.39
Fuel Inventory	110 ton			
Uranium cost	$40/kg	4	4	0.11
Fabrication	$30/kg	3	4	0.09
Fuel consumption	25.8 ton/yr @ 9,000 Mwd/ton			1.17
Operating costs†				0.72
TOTAL		62		5.27

* Prepared by W. B. Lewis for D_2O-Zr-UO_2 design.
† Includes 1 lb/hr D_2O loss.

reactors planned by well-qualified, independent groups is perhaps the best indication of the usefulness of enrichment, and the confidence of reactor designers that such fuel will be available for a long time."

Shoupp was even more emphatic: "The developmental advantages favor the stainless-steel–H_2O–enriched-uranium case and I believe the possibility of achieving low costs is greater from a technical standpoint for this reactor system. The developmental and operational problems will probably be simpler and easier to overcome for enriched reactors and, on the general concept of enriched vs nonenriched reactors, the exclusive use of natural uranium would rule out many promising types such as single and two-region homogeneous reactors, seed cores, fast breeders and stainless-steel-clad–H_2O multiregion reactors."

Natural-uranium's rebuttal. The essence of the rebuttal for unenriched reactors was that the estimated power costs for the natural-uranium designs now being considered look hard to beat. It may be a struggle for the enriched reactors just to match these rates much less surpass them.

Lewis (Atomic Energy of Canada Ltd.) presented cost figures for an 800-Mw (thermal) D_2O-Zr-UO_2 design based on Chalk River experience. Using prices quoted by the U. S. AEC for fuel and D_2O, he calculated a total power cost for a reactor operating in Canada of 5 mills/kwh. A breakdown of this cost is in the table above.

Lewis's key assumption was that the natural-uranium fuel elements can be irradiated to 9,000 Mwd/ton. Reactivity is maintained by irradiating the fuel in a continuous fashion so that the fresh elements compensate for the reactivity loss in spent elements.

Menke also gave the power cost estimates for a 200-Mw (electric) version of the sodium-deuterium reactor that is being designed by NDA for the Chugach Electric Association in Alaska. Menke expressed these views concerning the design: "In this 200-Mw high-temperature, heavy-water-moderated, natural-uranium plant we believe that the reactor, turbogenerator and buildings all together to the bus bar would amount to a capital investment of $310/kw (electric).

"We believe that power can be removed from this complex at a cost of about 8.4 mills, based on 13% capital cost, 80% load factor, 1.6 mills fuel components in that cost, burning to 5,000 Mwd/ton and crediting the plant with only 25% efficiency, which we believe can be beaten, as we believe the 5,000 Mwd eventually will be beaten, and taking no credit whatever for the plutonium in the spent elements."

Conclusions

To the observer of this debate it seems that enriched reactors must rely heavily upon their development potential to do much better than match the power costs of natural-uranium systems.

The final result may very likely be that individual circumstances will determine which reactor is used in a particular situation. Countries like the United States that have high interest rates and lots of enriched uranium may find that the slightly enriched reactor is best while other countries that have low interest rates and no enriching plants may, with just as much reason, decide in favor of the unenriched systems. Starr's last words on the subject were perhaps most appropriate:

"We will not know for some time what the trend will be. Even in the future, different circumstances may demand different types of reactors. What we need now is to continue to push ahead on all fronts so that all the reactor concepts—enriched or natural—can prove their merits in actual operation."

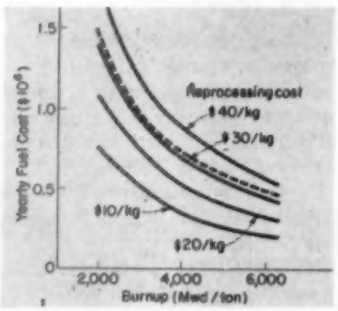


FIG. 2. Fuel costs for a large power reactor assuming reprocessing (solid lines) and no reprocessing (dotted line). Pu produced is credited with a buy-back value of $12/gm

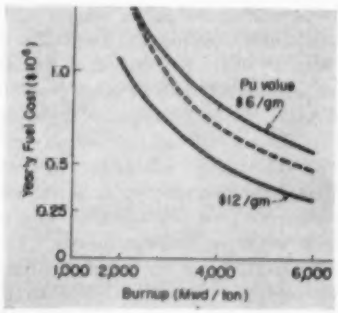


FIG. 3. Fuel costs for a large power reactor assuming reprocessing (solid lines) and no reprocessing (dotted line). $20/kg has been charged for fuel reprocessing

PWR and Calder Hall—How Do They Compare?

Depending on exchange rates, labor costs, a country's know-how and other complex factors, different reactors may be best in different situations. This analysis studies the big two

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MANY NATIONS, particularly those in Europe and Asia, are trying to answer one of the most puzzling of questions: Which type of nuclear reactor has most promise for economic atomic power?

Reactor comparison is a complex task. Many technical, political, and economic factors enter that can be quite dependent on the particular situation existing in a country. It is our purpose to examine the most important of these factors and as an example to illustrate their influence on the power-generating costs of the Calder Hall reactor type and the PWR reactor type for construction in the United States and for construction in Great Britain.

Significant changes are wrought by the economic assumptions used in comparing two different reactor types. Thus before deciding one type is superior to another, careful consideration must be given to such questions as: What is the true cost of capital needed to construct the reactor? What are the potential improvements in the respective types? Which type will be more attractive under the projected economic conditions (cost of money, foreign exchange, taxes, etc.) likely to prevail in the long run? Which will best serve the national interests of the country concerned? Is the reactor to be constructed as the best presently possible power producer, or as a pilot operation aimed at even lower cost power in the somewhat longer range? When decisions have been made on these factors, a more realistic comparison of reactor types can be achieved.

Table 1 is designed to help answer this question. It presents the economics of PWR and Calder Hall reactor types under two different conditions:

1. Construction of both reactors in the U. S. using U. S. private-utility accounting practices.

2. Construction of both reactors in Britain using capital charges based on British Electricity Authority accounting practices.

Technical Factors

Certain technical features of reactor design can have a decided influence on the best choice of power reactor for a nation—for example, the know-how of a country in the fabrication of specialized equipment such as large pressure vessels. As small an item as the clearance along a railroad right-of-way can determine whether a large pressure vessel can be shop-erected or must be field-erected, with corresponding limitations on possible equipment design. The past experience of the country in production-type reactors will influence the course of power-reactor development. Thus the British have operated gas-cooled production reactors and favor the gas-cooled approach to atomic power while the U. S. has chosen water-cooled production reactors and concentrated more on this type for power generation.

Assumed improvements. Another important technical factor that influences a choice is the degree of extrapolation from present technology that the evaluator assumes. If one is willing to look far enough into the future, almost anything appears possible. Based on fairly reasonable assumptions, it is likely that atomic power costs will eventually achieve the 7-mill range. However, such low costs seem 15–20 years away under U. S. accounting practices, in which fixed charges play an important part in determining power cost. During this period of cost reduction in atomic power, there may well be shifts in the relative attractiveness of reactor types, and it seems logical to talk about "short-term winners" and "long-term winners." Thus in any particular comparison of types it is important to state specifically just what the assumptions are with regard to time period and state of technology.

The areas of nuclear power costs in which substantial reductions appear quite probable are:

Heat flux. The bottleneck in the output from most reactor cores is the allowable heat flux. In early reactor designs, it has been necessary to be quite conservative in choosing design values for heat fluxes because of uncertainty as to corrosion resistance of fuel

cladding, distribution of coolant flow in the reactor, proper hot-channel factor to be employed, etc. A substantial improvement in reactor heat output seems probable as operating experience is obtained in power reactors.

Radioactivity. Conservative design practices have also been followed with respect to radioactivity hazards. This leads to interposition of intermediate loops and additional heat exchangers to avoid possible radioactive contamination of the power-cycle fluid and addition of spare equipment and elaborate instrumentation and shielding. With operating experience, it should be feasible to eliminate many of these precautions.

Fuel costs. Fuel-fabrication and fuel-processing costs should show continual and significant reduction by the achievement of longer exposure levels for each element and by economies in manufacturing and chemical operations in the fuel cycle.

Political Factors

The choice of reactor type is also affected by considerations that are primarily political rather than economic. Thus even though enriched uranium can be purchased from the United States or Britain, some countries lacking the capability of enriching uranium themselves may favor the natural-uranium reactor on the basis that it employs a more widely available fuel.

The relative urgency of power and fuel requirements differs greatly in various countries. In countries expecting high power costs from conventional fuels, this encourages the construction of the reactor that is most economical right now. In the United States low power costs permit construction of a type that holds the most promise for economical operation in the future.

Economic Factors

It is difficult to generalize regarding the over-all economics of nuclear power from reactor types because the economic results tend to be very sensitive to a number of the factors underlying the annual fixed and operating charges for power generation.

Construction costs. There are great differences between nations with respect to equipment costs, building materials, labor efficiencies, labor rates, overhead rates, etc. The cost of land and the necessity for exclusion areas can also vary. These differences can have a major effect on the cost of building a reactor in one country versus another. An additional complication is the variety of exchange rates in controlled-currency countries. This can have a significant effect on the cost of imported equipment. The cost of such equipment often depends on the decision of a currency-control board rather than on any "market price."

TABLE 1—Power Costs for 90,000-kw Calder Hall and PWR

	Cost of construction of reactor in U. S. (mills/kwh)*		*Cost of construction of reactor in Britain* (mills/kwh)*	
	Calder Hall	*PWR*	*Calder Hall*	*PWR*
Invested capital				
Construction	12.8	7.4	4.8	2.8
Capital charges during construction	1.2	0.8		
Start-up	0.1	0.1		
Investment in first core	0.3	1.1	0.1	0.4
Total	**14.4**	**9.4**	**4.9**	**3.2**
Fuel costs				
U^{235} burnup	1.4	2.0	1.4	2.0
Plutonium credit	1.4	0.5	1.4	0.5
Net burnup	**0.0**	**1.5**	**0.0**	**1.5**
Rental of fuel	0.4	0.2	0.5	0.2
Fuel fabrication and processing	1.6	7.5	1.6	7.5
Total	**2.0**	**9.2**	**2.1**	**9.2**
Site operation	1.5	1.0	1.0	0.7
Power cost	**17.9**	**19.6**	**8.0**	**13.1**

* All costs in mills/kwh. For U. S. construction capital charges are based on U. S. private-utility accounting practices. For Britain capital charges are based on British Electricity Authority accounting practices.

There are at least three methods of attempting to translate the construction cost of a reactor design, prepared in one country under that country's design practices and cost situation, to the construction cost of the same reactor in another country:

1. A weighted construction-cost index can be based on the relative costs of similar types of plants in the two countries. This can be used for converting a construction cost in one country to that in the second.

2. Based on the design specifications of the plant to the extent consistent with design and construction practices of the second country, a direct construction-cost estimate can be made for building the plant in the second country.

3. A general over-all correction factor can be estimated from the experience of firms that have built similar plants in both countries and have reached conclusions as to the relative costs over and above the simple rate of exchange.

Fixed-cost assessment. Plant amortization, return to investors, bond interest, taxes, and insurance are the principal items that must be applied to the construction cost of a nuclear power plant to calculate the annual charge to be levied against the power produced each year.

The annual fixed-charge rate can vary widely, depending on whether a plant is privately financed or financed

by tax-free, low-interest government funds (*1*). An annual charge without amortization or taxes can range between 2.25% and 9.9%, depending on whether financing is by the U. S. Government or by a private U. S. utility. Since nuclear plants at present have high capital costs, which may comprise as much as 75% of the total cost of the power generated, the type of financing used on these plants will exert a strong leverage on the calculated power cost.

Thus when trying to decide between two reactor types for even one country, it could very well be a logical conclusion that Type A would be better if the government financed the plant, while Type B would be better if a private utility were to finance the plant.

An additional consideration that often can be misleading is the possible availability of low-interest money from such sources as the World Bank for erection of early nuclear power plants in some foreign countries that have high interest rates. Questions inevitably arise as to the "real" cost of such a loan in terms of other projects that are ruled out by the allocation for the nuclear plant of a portion of the available low-interest money.

Fuel-cycle costs. Assumptions as to the state of technology will have an important effect on fuel-cycle economics. The best choice of reactor will often depend significantly on degree of optimism and assumption of short-term or long-term technological improvements. For example, zirconium cladding for the fuel in water-cooled reactors will permit significant advances in neutron economy but is presently much more expensive than stainless steel. However, there has been a recent cost reduction by a factor of two in zirconium sponge, and steady improvements are being made in the yield of acceptable fabricated zirconium parts. The establishment of fairly large-scale plants for handling slightly enriched U^{235} and removing plutonium from slightly enriched fuel elements should permit significant cost reductions compared with the present expense in plants not specifically designed for such material.

Design Comparison

The effect of location and technical, political, and economic considerations is well illustrated by comparing the power-generating economics of Calder Hall and PWR types. In many reactor comparisons, it is sufficient to use consistent assumptions with respect to such items as construction costs, capital charges, state of technology, etc. However, when Reactor A has a high capital cost and low operating cost, while Reactor B has a low capital cost and high operating cost, consistency alone is insufficient, since the two designs have their economic strengths and weaknesses in different areas. Levying a high capital charge, for example, will penalize one design much more severely than the other.

This is particularly true in evaluating PWR and Calder Hall. Calder Hall is a deliberately simple, conservative, low-pressure reactor, characterized by Sir Christopher Hinton as "the slow-speed reciprocating engine of the reactor world." It has a high construction cost and a low fuel-cycle cost. PWR is a high-pressure reactor with an expensive, watchmaker's fuel element. It has a lower construction cost, but its fuel-cycle cost is high because of the costly zirconium fuel elements. Either design could be made more economic, for example, by the proper choice of capital charge.

Plant size. In nuclear power plants, the unit fixed charges for power will fall rapidly as reactor size increases. Hence for strict comparability two reactor types should be compared at the same power output. In this particular study an output of ~100,000 net electrical kw was of interest. Since technical and economic details are available for a PWR design of 90,000 electrical kw, which has been described as feasible for construction now (*2*), this output was selected as the basis for comparison.

The advanced Calder Hall plants to be built in Britain consist of two-reactor stations with outputs from each reactor of 130,000–160,000 net electrical kilowatts and capital costs of $350/kw–$310/kw, based on construction in Britain. It seems to be a fairly good rule that over a reasonable size range the cost per kilowatt will vary about as $(\text{power})^{-\frac{1}{2}}$. By this rule, the cost of a 90,000-kw Calder Hall reactor constructed in Britain would probably be about $425/kw.

Differences in construction costs between the U. S. and Britain have been given careful attention. Our procedure was to examine the major components of construction costs and to determine relative costs between the two countries for each of these components. Table 2 shows the results. With the information available, it appears that a given design would cost about 50% more to construct in the U. S. than in Great Britain.

This conclusion was checked independently by making a direct cost estimate of a 90,000-electrical-kw Calder Hall reactor for U. S. construction. A total of about $630/kw for such a plant in the U. S. resulted from both this direct cost estimate and multiplying the cost in Britain by 1.5.

TABLE 2—Relative Construction Costs—U. S. vs Britain

	A *Percentage of total for each item in U. S. construction practice*	*B* *Ratio (British cost/ U. S. cost)*	*C* *Equiv. British cost factor (A × B)*	*D* *Equiv. British percentage of total*	*E* *Ratio (U. S. cost/ British cost)*	*F* *Equiv. U. S. cost factor (D × E)*
Major items large equipment	5	1.33	0.066	5	0.75	0.037
Major items small equipment	20	0.83	0.166	25	1.20	0.300
Building materials	25	0.91	0.227	30	1.11	0.333
Construction labor*	35	0.40	0.140	25	2.50	0.625
Construction overhead	15	0.50	0.075	15	2.0	0.300
Totals	100		0.674	100		1.595

* Corrected for 50% productivity of British labor compared with American labor.

For construction of the PWR in Britain, the cost of $31.1 million—estimated here in the U. S. for a "second-generation" Shippingport design—was divided by 1.5 to give the approximate cost of construction of this design in Britain. Unit costs for fuel-cycle steps for each reactor were assumed to be the same in Britain and the U. S. Information on the Calder Hall fuel-cycle costs in Britain is meager.

An attainable exposure level of 3,000 Mwd was assumed for the Calder Hall fuel element. This is based on British estimates. Reactivity using natural-uranium fuel would probably limit the exposure level to 5,000–6,000 Mwd/ton, in any event. An exposure level of 6,000 Mwd/ton was assumed for the PWR natural-uranium-oxide blanket. Because of the use of oxide, radiation damage probably should not limit the exposure below this level. It was assumed that sufficient PWR seed cores per blanket core were used to maintain reactivity.

Cost basis. Costs in Table 1 were based on the following annual charges on amortizable items:

	U. S.	*Britain*
Reactor	15%	9%
Turbogenerator plant	15%	8%
Other fixed costs	12%	8%
Capital charge for fuel outstanding	4%*	5%

Conclusions

Thus in Britain, it appears that nuclear power from the Calder Hall reactor can indeed be produced at about 8 mills, which makes it competitive with new conventional plants in Britain using coal at the high marginal cost of this fuel with which the British are now faced. Because the Calder Hall reactor is a high-capital-charge, low-operating-cost design, it benefits more from low-cost financing and appears substantially better than the PWR in Britain. In the U. S., however, the higher capital charges are a more severe penalty to the Calder Hall design, and the two reactor types would produce power at roughly 18–20 mills/kwh.

The following points are significant in this comparison:

1. For given reactor designs, the construction-cost level and the type of financing has a profound effect on relative economics. In such second-generation designs, the capital charge is important, and the low-cost financing assumed in Britain is of substantial benefit, particularly in the economics of the Calder Hall reactor.

2. Because of the high capital cost of nuclear power plants today and the severe economic penalty placed on high capital costs by the fixed-charge rates characteristic of U. S. private-utility accounting practices, atomic power from either of these two designs is a long way from being economic in the U. S. at present. However, if reactors of larger output had been assumed, the unit power cost would have been considerably less. The expected improvement in core heat output, fuel-element life, and fabrication and processing costs should also in time permit additional cost reductions.*

Advanced Designs

When one looks beyond the designs compared here, it appears that the PWR type of reactor has somewhat more potential for improvement than the Calder Hall design. In PWR, for example, the maximum heat flux is reported to be about 380,000 Btu/ft²/hr. In less conservative designs of pressurized-water reactors, such as the AEC-sponsored study, "Project Dynamo" (*3*), it was felt that ultimately the attainment of considerably higher heat fluxes should be possible, even including a 30% safety factor. These are the heat fluxes in Btu/ft²/hr:

	PWR Shippingport	*PWR Dynamo*
Maximum	380,000	1,000,000
Average, blanket	65,000	270,000
seed	112,000	270,000

In the Calder Hall design, both the magnesium-cladding temperature and the uranium-metal center temperature are pushing very close to the tolerable limits. Hence improvements in Calder Hall power output would seem inevitably to involve a major redesign of the fuel element. Since the present element is extremely inexpensive, it is questionable if a net economic saving of any magnitude could be effected by such redesign. Similarly with the use of slightly enriched U^{235}, of helium or hydrogen as the coolant instead of carbon dioxide, of beryllium instead of magnesium as the fuel cladding, the potential for cost reduction in Calder Hall does not seem as great as for the pressurized-water type.

Whither Other Nations?

A similar approach to the comparison of Calder Hall and PWR in Britain and in the U. S. can be used to determine the merits of these reactor types in other countries. Again the construction-cost index and the fixed-charge rate are the most important variables. In addition some extra cost can be involved in shipping spent fuel elements to the U. S. or Britain for plutonium removal.

Such a study was conducted for location of these two reactor types in Japan. There the interest rate is even higher than in the U. S., but taxes are lower. On balance it appears that the fixed-charge rate, assuming financing at current Japanese money rates, would be about the same as in the U. S. (15%). Construction costs for an atomic plant in Japan, assuming much of the equipment would have to be imported, should run about the same as in the U. S. In general, the plant factors for power plants in Japan tend to be somewhat lower than in the U. S. and Britain, and it was assumed that this would continue to be true for nuclear plants. By such ground rules, the estimated power cost in Japan for a 90,000-kw PWR was found to be 21.8 mills/kwh with a corresponding cost of 21.0 mills for the Calder Hall reactor of the same size.

* * *

This article has been abstracted from the American-Standard report, "Comparison of Calder Hall and PWR Reactor Types," performed under Contract No. AT(04-3)-100 for the Atomic Energy Commission, Division of Reactor Development. Copies of this report are available from the Office of Technical Services: Report No. AECU 3308.

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* This is the rental charge decided on by the USAEC for fuel material. For Britain it is assumed that such material is purchased rather than rented, with the average interest rate of 5% on funds for such purchase.

* It is important to keep in mind that a principal reason for the decision to construct the Calder Hall and PWR reactor types was the technical assurance that such designs would work. Preliminary studies of other more advanced reactors indicate that at least on a design basis these advanced types should produce power at appreciably lower costs than either Calder Hall or PWR.

A First Look at

Fusion-Reactor Control

A "black-box" approach to the fusion reactor predicts over-all control characteristics resembling those of a conventional fission reactor

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THE FUSION REACTOR concept has been developed to the point where there is now enough information to enable discussion of some of the associated engineering problems. For instance, R. F. Post's article "Controlled Fusion Research" (*1*) provides an excellent background for speculation as to what a controlled fusion reactor might eventually look like and how it might be expected to behave. Although the basic conceptual problems outlined by Post appear to be most formidable, it is quite clear that the succeeding engineering problems will easily match them in complexity.

Attempts to understand these problems can begin by comparing some of the problems with currently understood fission-reactor problems. The reactor control engineer in particular can begin to be interested in questions of stability, control, and speed of response.

Reactor stability, in the mind of the control engineer, concerns the reactor as a whole and as a component in the complete plant. Detailed effects, such as neutron-flux spatial variations in the fission reactor or plasma oscillations in the fusion reactor, are of little concern. Problems of plasma instabilities that affect the basic feasibility of a fusion-reactor design [for example, the type predicted by Kruskal and Schwarzchild (*2*)] will presumably have been solved before the engineer is required to design a control and protective system.

These considerations lead to the "black-box" approach to over-all reactor stability in which no geometry need be specified and the performance of the "box" as a controlled power generator is given by kinetic equations in terms of a few variables.

Fission-Reactor Example

The approach used to analyze the kinetic behavior of a conventional fission reactor is a familiar example of the black-box method. The "black-box" picture of a simple fission reactor is shown in Fig. 1. The kinetic performance of such a reactor without temperature coefficient is mathematically described by the following well-known equations:

$$\frac{dn}{dt} = \frac{\delta k}{l^*} n - \frac{\beta}{l^*} n + \sum_{i=1}^{6} \lambda_i C_i + S \quad (1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} n - \lambda_i C_i \quad (2)$$

The symbols are conventional (*3, 4*).

FIG. 1. Black-box representation of fission-reactor control problem. k is reactivity, n is neutron flux and S is neutron source

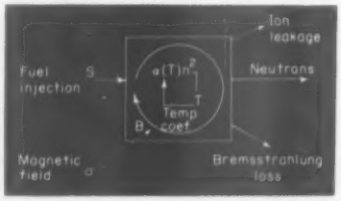

FIG. 2. Black-box representation of fusion-reactor control problem. $\alpha(T)n^2$ is reaction rate, T is reaction temperature, B is magnetic field and S is fuel injection rate

Equation 1 is a neutron-balance equation, and Eq. 2 describes the time behavior of delayed emitter concentrations. It will be noted that there is no concern about time variations in the spatial distributions of neutron flux or emitter concentrations. The equations are concerned rather with how the entire box responds as a unit with time. From these equations the frequency response or transfer function, relating output to input, can be derived. An examination of the transfer function then allows prediction about the stability of the reactor and gives insight into the general characteristics required of an external control system. This gross handling of a complex, detailed subject has been most successful for the fission reactor and consequently leads one to attempt a similar approach for the fusion reactor (see Fig. 3).

Fusion-Reactor Black Box

The corresponding "black box" for a fusion reactor in which the reaction is presumed to be confined by a magnetic field might be that of Fig. 2. Several other configurations of black boxes can also be made, but at this stage it seems more important to illustrate the approach rather than present all of the specific concepts. The equations and constants describing this box follow Post's notation and graphs.

Two equations of roughly the same form as the fission equations can be written for a simple confined plasma system:

$$\frac{dE}{dt} = \tfrac{1}{2} n^2 \langle \sigma v \rangle_{\alpha+\alpha} W - 0.54 \times 10^{-30} n^2 T^{1/2} - \frac{K_D n k T}{\tau(T)} \quad (3)$$

$$\frac{dn}{dt} = -\frac{K_{D_1} n}{\tau(T)} - \tfrac{1}{2} n^2 \langle \sigma v \rangle_{av_8} + S \quad (4)$$

Equation 3 is a power-balance equation that represents the rate of change of internal energy in the plasma, and Eq. 4 represents a particle-number balance. Since all that follows is concerned entirely with obtaining and interpreting the mathematical solution of these two equations, an attempt will first be made to describe, and hopefully to justify, the terms of the two equations and the many assumptions that have been made.

The assumption is made that some suitable configuration has been found for a fusion reactor, and this reactor will be first considered as operating on the DD reaction. None of the generalizations to be made is affected when the DT reaction is used. The deuterium ions in the "box" have a density n, are at an elevated temperature T, and are presumed to be confined in a nonspecified manner by means of a fixed constant magnetic field B. The process is assumed to be occurring on a per-unit-volume basis, and the over-all volume and geometry are ignored. As in the case of the fission-reactor equations, the variables can also represent either total or average values for the entire box.

dE/dt is the rate of change of energy within the "box" and can be represented by the rate of change of the thermal energy possessed by the system; hence, $dE/dt = d(\frac{3}{2}nkT)/dt$, where k = Boltzmann's constant. The power balance consists of two parts: first, a production term representing thermonuclear energy release, and, secondly, various losses of power by specific processes. The production term is $\frac{1}{2} n^2 \langle \sigma v \rangle_{av_8} W$, where the constant W is the energy released to the charged particles per reaction. The energy possessed by the neutrons generated in the DD reaction is presumed to be lost from the system, gainfully or otherwise, and therefore is not available to keep the reaction going. The $\langle \sigma v \rangle_{av_8}$ term, the average of the product of cross section and relative velocity as given in Fig. 4, is dependent on temperature and is designated hereafter as $\alpha(T)$.

The first loss term of Eq. 3 is given by $0.54 \times 10^{-30} n^2 T^{1/2}$. This is the Bremsstrahlung radiation loss from the system. Here the electrons radiating are assumed to be in temperature equilibrium with the ions (the electron particle density, equals the ion particle density).

The second loss term $K_{D_1} nkT/\tau(T)$ is the power loss from the system by particle diffusion, where $\tau(T)$ represents the average time taken by particles beginning at the source to drift through the magnetic field to the walls of the box. This "confinement time" must be comparable to the average reaction time for a successful fusion reactor.

Other loss terms can also be imagined, such as the radiative losses caused by impurities and the possible need to provide internal power to heat up new particles inserted into the system. These other losses only add additional negative terms to Eq. 3 and are a refinement that does not alter substantially the form or the approach to the control-stability problem.

In Eq. 4 the term $K_{D_1} n/\tau(T)$ can be thought of in fission-reactor terms as the particle loss caused by leakage. Similarly, the second term, $\frac{1}{2} n^2 \langle \sigma v \rangle_{av_8}$ is analogous to fuel depletion or burnup. The third term is a source or injection term which refers to the rate of addition of new particles to the system to compensate for the burnup and leakage.

It will be noted that the roles of the source term in the fission and fusion reactors are somewhat reversed. In a fission reactor as the power level is increased the effect of the source becomes smaller and smaller. In the fusion reactor at higher power levels more and more particles are needed to keep the reaction going. Injection is probably a better term for use with fusion reactors in that the introduction of new fuel is actually implied.

Mathematical Solution

We must first consider the form of the confinement time $\tau(T)$ with respect to the temperature and the magnetic field. The confinement time is inversely proportional to an average drift velocity. Post gives two forms for this velocity—the so-called "classical" diffusion theory in which $\tau(T) \sim B^2 T^{1/2}/n$ and Bohm (δ) diffusion theory where $\tau(T) \sim B/T$. For our purposes either form is suitable, and we first select Bohm diffusion. Let

$$\tau(T) = \eta \frac{B}{T}$$

where η is an appropriate constant. An indication will be given later as to the effect upon stability of changing the form of the diffusion equation.

We can now rewrite Eqs. 3 and 4 in terms of the auxiliary constants defined in the table as follows:

$$K_1 \frac{d}{dt} nT = K_2 n^2 \alpha(T) - K_B n^2 T^{1/2} - K_D n T^2 \quad (3a)$$

$$\frac{dn}{dt} = -\frac{2}{3}\frac{K_D}{k} nT - \tfrac{1}{2} n^2 \alpha(T) + S \quad (4a)$$

These equations are nonlinear, but as in the case of the fission reactor useful solutions are obtained by linearizing them. The object here is to derive a transfer function expressing the re-

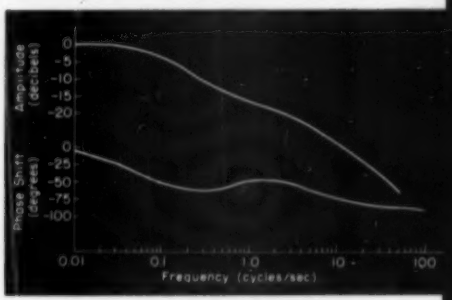

FIG. 3. Frequency dependence of transfer function $\Delta n/\Delta S$ relating time variation of ion density n and injection rate S for a fusion reactor operating on the DT reaction

Definitions of Auxiliary Constants

Auxiliary constant	*Definition*	
	DD reaction	*DT reaction*
K_1	$\frac{3}{2}k$	$\frac{3}{2}k$
K_2	$\frac{1}{2}W$	$\frac{1}{4}W$
K_D	$\frac{K_{D_1} k}{\eta B}$ (Bohm)	Same as DD
A	$K_2 K_\alpha n_0^2 - \frac{1}{2} K_B n_0^2 T_0^{-1/2} - 2K_D n_0 T_0$	Same as DD
B	$2n_0\alpha_0 K_2 - 2K_B n_0 T_0^{1/2} - K_D T_0^2$	Same as DD
C	$\frac{2}{3}\frac{K_D}{k} n_0 - \frac{1}{2} K_\alpha n_0^2$	$-\frac{2}{3}\frac{K_D}{k} n_0 - \frac{1}{4} K_\alpha n_0^2$
D	$-\frac{2}{3}\frac{K_D}{k} T_0 - n_0\alpha_0$	$-\frac{2}{3}\frac{K_D}{k} T_0 - \frac{1}{2} n_0\alpha_0$
E	$K_1 n_0$	Same as DD
F	$K_1 T_0$	Same as DD

sponse of the dependent variables T and n to sinusoidal variations of the injection term S. The characteristic equation so determined will then indicate whether the system is stable and will show the effect of the various parameters on stability. The process consists in assuming that each of the time variables, n, T, and S, is made up of a steady-state value and a small sinusoidal variation about the steady-state value:

$$n = n_0 + \Delta n(t)$$
$$T = T_0 + \Delta T(t)$$
$$S = S_0 + \Delta S(t)$$

The variable coefficient $\alpha(T)$ also is linearized as

$$\alpha(T) = \alpha_0 + K_\alpha \Delta T$$

We neglect second-order differentials and note that a new term K_α is introduced which is analogous to the temperature coefficient of reactivity in a fission reactor. These new variables can now be introduced into Eqs. 3a and 4a. With new constants, as defined in the table, we have finally

$$E\frac{d\,\Delta T}{dt} + F\frac{d\,\Delta n}{dt} = A\,\Delta T + B\,\Delta n \quad (5)$$

$$\frac{d\,\Delta n}{dt} = C\,\Delta T + D\,\Delta n + \Delta S \quad (6)$$

These equations are now linear with constant coefficients; so we can take the Laplace transforms. The zero conditions are ignored as we have already subtracted the steady-state terms.

$$Es\,\Delta T(s) + Fs\,\Delta n(s) = A\,\Delta T(s) + B\,\Delta n(s) \quad (7)$$

$$s\Delta n(s) = C\,\Delta T(s) + D\,\Delta n(s) + \Delta S(s) \quad (8)$$

The transfer functions for Δn and ΔT can now be obtained directly using $\Delta S(s)$ as a forcing function.

$$\frac{\Delta T(s)}{\Delta S(s)} = \frac{B - Fs}{Es^2 + (FC - ED - A)s + AD - BC} \quad (9)$$

$$\frac{\Delta n(s)}{\Delta S(s)} = \frac{Es - A}{Es^2 + (FC - ED - A)s + AD - BC} \quad (10)$$

These transfer functions, which are of the form

$$\frac{s + c}{(s + a)(s + b)}$$

start off with a finite gain at zero frequency. They have three break points, assuming a and b are real, and the amplitude response ultimately falls off at a rate of 6 db per octave. The roots can very easily be obtained once specific numerical values are used.

Stability Criteria

We now are in a position to examine the stability of the system. The system is stable if the roots of the characteristic equation (the denominator of Eqs. 9 and 10) have negative real parts. A necessary and sufficient condition for the roots of a quadratic equation to have negative real parts is that all of the coefficients have the same sign. As we know that the energy E must be positive, to have a stable system it is necessary that

$$FC - ED - A > 0 \quad (11)$$
$$AD - BC > 0 \quad (12)$$

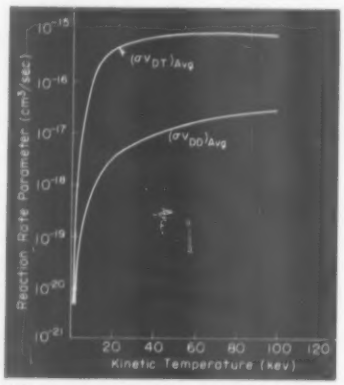

FIG. 4. Temperature dependence of reaction-rate parameter for DT and DD reactions assuming Maxwellian particle distribution. This figure from article by R. F. Post (1)

We can now see what these criteria mean in terms of the original constants. First using inequality 11 and substituting the expressions from the table, we find that, for stability,

$$\tfrac{3}{2}k\alpha_0 - \tfrac{3}{4}kK_\alpha T_0 > \tfrac{1}{2}K_\alpha W - \tfrac{1}{2}K_B T_0^{-1/2} - \frac{2K_D T_0}{n_0} \quad (13)$$

This inequality reveals several interesting bits of information. It will be noted that the terms to the right of the inequality are related directly to the gain and loss terms of Eq. 3 with the last two terms representing the losses. From the negative signs it appears as though the greater the losses the higher the possibility of stability. Physically this seems reasonable, as a system with large losses is heavily damped and not capable of continuous oscillations.

Other inferences can be drawn from this inequality by examination of K_α, the temperature coefficient. If K_α is negative, meaning a decreased reaction rate with an increase in temperature, the system is clearly stable, since the other pertinent constants are all positive. (From Fig. 4, $\langle\sigma v\rangle$ for the DT reaction shows a negative slope over a portion of its range. The DD reaction does not exhibit this characteristic in the indicated range.) If K_α is zero or slightly positive, the system also appears to be stable. The maximum positive value permissible for K_α depends on T_0, n_0, and also K_D, which is a function of the confinement process and field strength. Without considerable refinement and good numerical data the above statements can be interpreted only as trends either toward or away from stability.

The remaining stability criterion can be obtained by substituting the original constants into inequality 12

$$2K_D T_0\left(\alpha_0 - \frac{K_2 K_\alpha}{3k}\right) + T_0^2 K_D\left(\frac{2}{3}\frac{K_D}{kn_0} - \tfrac{1}{2}K_\alpha\right) + \tfrac{1}{2}K_B\alpha_0 n_0 T_0^{-1/2} > K_B T_0^{1/2}\left(\frac{K_D}{k} + K_\alpha n_0\right) - \tfrac{5}{3}K_2\frac{K_D}{k}\alpha_0 \quad (14)$$

This inequality does not appear to have the obvious physical significance that inequality 13 does. However, if we assume that 13 imposes a lower bound on the temperature for stability, at a high temperature there seems to be a form of density limit from 14. It can be seen that at high temperatures and high densities the $K_B T_0^{1/2} K_\alpha n_0$ term might ultimately predominate if K_α remains positive and finite. Stable operation then appears to have a low-temperature limit and a high-density limit at high temperature.

To indicate that the trend remains the same regardless of the form of the diffusion, we can again solve Eqs. 3 and 4 using the classical type of diffusion rather than the Bohm type. The ine-

quality corresponding to 13 becomes

$$\tfrac{3}{2}k\alpha_0 - \tfrac{3}{4}kK_\alpha T_0 > \tfrac{1}{2}K_\alpha W - \left(\frac{K_B + K_D}{2} + \tfrac{5}{2}K_D\right) T_0^{-\frac{1}{2}} \quad (13a)$$

It is apparent that the diffusion has affected only the last term of the equation and all of the mentioned trends still are valid.

Frequency Response

Returning now to the transfer function, it is of interest to obtain order-of-magnitude answers for the frequency response of the reactor to get a feel for the control problem. For this step one must supply numerical values for the roots of the characteristic equation. As a convenient starting point toward obtaining numbers we will first try the DD reaction. Post's calculations (Fig. 4) suggest that fission reactors operate at a power density of about 100 watts/cm³. At this power density it is pointed out that a DD reactor operating at 100 kev kinetic temperature would need a particle density of 3.3×10^{15} particles/cm³. From these values the remaining constants can be obtained:

$$\begin{aligned} n_0^2K_B &= 6 \text{ watts/cm}^3\text{/kev}^{\frac{1}{2}} \\ K_2 &= 1.9 \times 10^{-13} \text{ watt-sec/reaction} \\ K_D &= 1.32 \times 10^{-19} \text{ watt/kev}^2 \\ K_\alpha &= 3.45 \times 10^{-19} \text{ cm}^3\text{/sec/kev} \\ \alpha_0 &= 3.05 \times 10^{-17} \text{ cm}^3\text{/sec} \\ k &= 1.6 \times 10^{-16} \text{ watt-sec/kev} \end{aligned}$$

K_D is obtained from the steady-state situation of Eq. 3 in which it is assumed that the diffusion losses just make up the difference between the production of the thermonuclear charged-particle energy and the Bremsstrahlung losses.

A check of inequality 13 indicates that the constants we have chosen represent an unstable system since

$$\tfrac{3}{2}k\alpha_0 - \tfrac{3}{4}kK_\alpha T_0 > \tfrac{1}{2}K_\alpha W - \tfrac{1}{2}K_B T_0^{-\frac{1}{2}} - \frac{2K_D T_0}{n_0}$$

$$7.32 \times 10^{-33} - 4.13 \times 10^{-33} > 65.5 \times 10^{-33} - 27 \times 10^{-33} - 8 \times 10^{-33}$$

It is evident that at the temperature and density chosen the positive temperature coefficient K_α is too large and quite dominates the stability picture. The depletion-loss term is small compared with the Bremsstrahlung losses, and it appears as though the conditions we have chosen do not provide sufficient damping to keep the system stable. We have the option of changing T_0 or n_0 in an effort to find better conditions. If, as is the case with the DT reaction, K_α ultimately becomes smaller or even negative as the temperature is further increased, we might hope to find stability at a higher temperature than 100 kev. Post's curves give no clue as to the magnitude of K_α for substantially higher temperatures. Attempts to improve the stability by lowering n_0 do not work within the limits of our assumptions. That is, with Bohm-type diffusion and the assumption that in the steady-state diffusion losses make up the difference between thermonuclear gain and Bremsstrahlung loss, n_0 and K_D vary directly with each other and hence a change in density does not affect the stability of inequality 13.

To continue our attempt to assign control speeds to the thermonuclear reactor, we will now derive the transfer function for a DT reactor operating in a range where we know the reaction has a negative temperature coefficient and where it must be stable. In this example for numerical values we can again assume the fission-reactor power density of 100 watts/cm³, a temperature of 100 kev, and an equal mixture of deuterium and tritium with the resulting total density of 4.2×10^{14} particles/cm³.

The basic stability equations are the same as Eqs. 3 and 4, but as shown by the table the constants are slightly different. The following values were derived from Post's curves:

$$\begin{aligned} \alpha_0 &= 8.2 \times 10^{-16} \text{ cm}^3\text{/sec} \\ K_\alpha &= -5 \times 10^{-18} \text{ cm}^3\text{/sec/kev} \\ W &= 5.75 \times 10^{-13} \text{ ev} \\ K &= \tfrac{1}{4}W = 1.44 \times 10^{-13} \text{ ev} \\ K_D &= 4.75 \times 10^{-18} \text{ watt/kev}^2 \\ A &= -0.531 \\ B &= 4.71 \times 10^{-14} \\ C &= -8.09 \times 10^{12} \\ D &= -2.15 \\ E &= 0.101 \\ F &= 2.4 \times 10^{-14} \end{aligned}$$

The transfer functions can now be obtained directly, and the density function becomes

$$\frac{\Delta n(s)}{\Delta S(s)} = 0.101 \frac{s + 5.27}{(s + 24.3)(s + 0.595)} \quad (15)$$

This transfer function is plotted in Fig. 3. It is evident that it is similar to the transfer function of a fission reactor. The last break point in the transfer function occurs at approximately 4 cycles/sec. In fission-reactor terms this would correspond to a reactor having a mean neutron lifetime of about 3×10^{-4} sec. This equivalent lifetime might cause the reactor to have a speed of response between that of a graphite-moderated fission reactor and a large water-moderated fission reactor, both of which are easily controllable.

Inasmuch as Fig. 3 shows a finite gain at zero frequency the fusion-reactor transfer function resembles that of a fission reactor having a negative temperature coefficient. (The transfer function for a fission reactor with zero temperature coefficient has infinite gain at zero frequency.) Thus all of the well-known properties of a fission reactor having a slight negative temperature coefficient would appear to carry over to this type of fusion reactor. The reactor in addition to being stable would provide a measure of self-protection against transients and would be incapable of "running away."

Conclusions

Many other interesting analogies between the two types of reactors could be derived from the equations presented here. It must be remembered, however, that the "black-box" approach must first be proved. Questions as to the effects of changing magnetic fields as well as the possibility of having moving boundary conditions have been ignored. Our attempt has been merely to show one type of engineering approach to an ultimate control problem.

However, in so far as our assumptions are valid, we have been able to establish two things: (1) because of its positive temperature coefficient the DD reaction will be unstable for temperatures up to at least 100 kev (about 10^9 °K); (2) the DT reaction, on the other hand, is stable below this temperature, and indications are that the control-system speed requirements would not be excessive.

* * *

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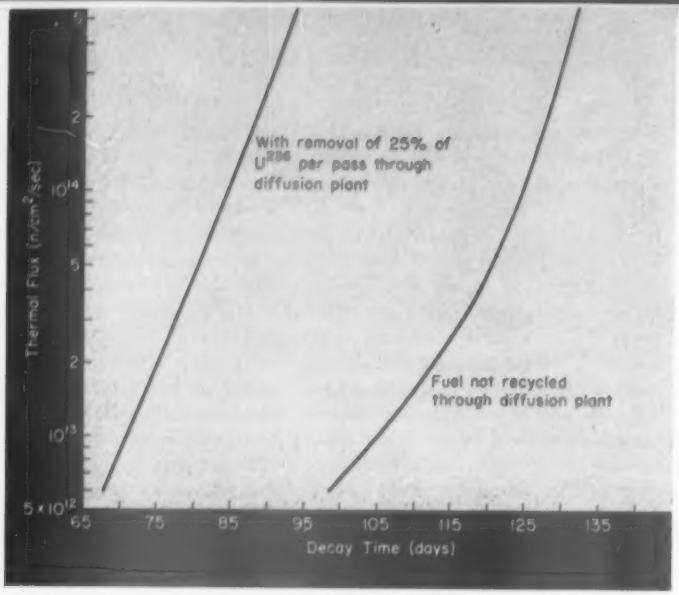

FIG. 1. Effect of U^{236} removal on decay time needed to reduce U^{237} to 10^3 dpm/kg uranium. Operating conditions: 1.2% U^{235} charged; $\phi t = 10^{21}$ n/cm²; batch irradiation; infinite recycle

Choosing Storage Time To Minimize Processing Costs

To determine the optimum fuel cycle, the cost of storing spent fuel must be balanced against processing, refabrication and shielding costs. These factors are considered here for various fuels and reactor types

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TO REPROCESS IRRADIATED FUEL, storage before or after chemical processing to reduce radioactivity, remote reprocessing methods, or a combination of these two approaches will be necessary. The final procedure will be determined by cost.

The cost of decay storage consists of three types of charges: (a) the cost of storage space, (b) the use charge for any fissionable material leased from the AEC, and (c) an inventory cost for working capital tied up in fissionable material bought from the AEC or awaiting sale to a customer.

To determine the total cost, it is necessary to define (a) the cost of the space and its operation, (b) the base charge for holding the fuel material (*1, 2*), and (c) how long the material will be held.

If material is leased, the basis for the use charge in most cases will be the value as taken from the AEC. If the material is purchased or awaiting sale, the working capital represented by process inventory should be calculated, based on its form at the particular moment in the processing cycle.

The decay time required depends on the radiation levels that can be tolerated for various refabrication techniques. Firm criteria have not been established to fix these levels. Since power reactors will discharge fuels that are highly irradiated by present standards, either long decay periods or a form of remote refabrication will be necessary. The economics of these alternates have not yet been fully investigated.

The length of the decay time should be based more upon the biological tolerances for the type of processing or refabrication necessary than upon arbitrary comparisons with natural uranium. The activity levels for semi-remote refabrication are assumed to be 30 mr/hr at 1 ft. This activity level delivers 1,200 mr per 40-hr week to the hands and 300 mr per 40-hr week to the total body, which is taken to be 2 ft from the source.

Irradiation Products

Reduction of fission-product activity after reactor discharge simplifies problems of heat removal during transport or storage and reduces radiation damage to process reagents and plant construction materials. Since 12.8-day Ba^{140} (La^{140}) controls shielding up to about 90 days' decay, considerable

TABLE 1—Proposed Handling of Transmutation Isotopes

Np^{239} in Pu^{239}	Allow 1-month decay of fuel, or recover Np^{239}
U^{237} in U^{235}	Allow 120-day decay of fuel, or increase refabrication cost
Pa^{233} in U^{233}	Allow 1-yr decay of fuel or waste, or recover Pa^{233}
U^{232} in U^{233}	Repurify periodically to remove U^{232} daughters
Th^{234} in Th^{232}	Allow 1-yr decay of fuel or product, or increase refabrication cost
Th^{228} in Th^{232}	Allow <60-day decay of fuel, or increase refabrication cost

savings in shield weight can be made by permitting decay before handling.

Finally, processing is simplified if there are fewer species of fission products to be removed from the fuel. For example, 8-day I^{131} can be eliminated from process consideration after about 100 days' decay.

Transmutation-product decay. Neptunium-239 is an intermediate in the production of Pu^{239} (see Fig. 2), and Pa^{233} is an intermediate in U^{233} production. Unless it is economically feasible to lose these unconverted intermediates, they must be either recovered or allowed to decay to the product isotopes.

Decay can be performed before the initial separation, or a second separation can be made after the waste from the first step has decayed. Since almost a year's decay is required to reduce the Pa^{233} content to 0.1% of its value at time of discharge, preprocessing decay may be ruled out as leading to excessive inventory charges. On the other hand, since only a month is required for comparable Np^{239} decay, it may be practical to allow decay before processing.

Uranium-237 and Th^{234} are produced in irradiations of U^{235} and Th^{232}, respectively. Product decay may be required since any undecayed U^{237} and Th^{234} will carry through the processing with the recovered uranium and thorium.

Seventy-year U^{232} produced from Th^{232} decays to 1.9-yr Th^{228} and its daughters. This leads to a particularly nasty situation—a short decay period will lead to U^{232} in the U^{233} product and some Th^{228} in the thorium product with subsequent ingrowth of activity into both; a longer decay period will increase the activity in the thorium product, which, since it is much larger in bulk than the U^{233}, does not help the situation. Processing after a short decay time to minimize the activity of the thorium product with periodic repurification of the uranium product has been proposed. Since such a scheme will feed appreciable quantities of Pa^{233} and Th^{234} to the process, storage of protactinium-bearing waste and product thorium may be required.

Buildup of Pu^{240} by continued recycle of plutonium may increase gamma and X-ray background by a factor of four over that of pure Pu^{239}. It also appears that the Pu^{238} content may have to be limited to keep the alpha-neutron reactions within the allowable neutron background.

Table 1 summarizes the proposed methods of transmutation-product handling.

Storage Space Required

The storage space necessary for irradiated-fuel decay is affected by fuel enrichment, processing-plant throughput, fuel-element geometry and alloy composition of the fuel.

Enrichment. As the concentration of fissionable material in an element increases, the number of elements stored per unit volume must be decreased to avoid creating a critical assembly. Against this can be balanced the fact that enriched elements will be shipped in smaller carriers, which can be handled with a lighter crane.

Plant throughput. Assuming a given type of fuel element with a fixed decay schedule, the area required for storage will be directly proportional to the plant throughput. In the case of enriched fuels, the cost of auxiliaries to maintain criticality control, such as baskets, will be roughly proportional to the throughput.

For a constant enrichment, over a wide range of plant throughput, the cost of the unloading area and heavy-duty crane will remain fixed. The operating costs, chiefly for personnel and water treatment, should not vary greatly with throughput.

Element geometry. For natural uranium, where criticality hazards are not considered for storage, the sole effect of fuel-element geometry is to change the weight that can be stored per unit volume.

In the case of enriched-uranium-alloy rods, the water-moderated critical mass is lower than the water-reflected one. A greater number of rods can therefore be stored tightly packed than that corresponding to the minimum possible critical mass.

Alloy composition. Any dilution of natural uranium by alloying constituents will have the same effect on storage cost as voids in the element, namely, increased cost per unit weight of uranium stored.

For highly enriched alloys, the situation is more complex. A safe procedure would be to assume that as the percent uranium in the alloy changes, the same mass of uranium may be stored in any one container. Therefore, more dilute alloys cost more to store than more concentrated alloys. Against this may be balanced the fact that in a more dilute alloy more poison will be present and more neutron leakage will occur, and therefore it might be permissible to store a larger mass of uranium per container. The particular range of moderator-to-uranium

TABLE 2—Comparison of Two Reactor Types

	One-region reactor	*Two-region reactor*
Enrichment	2%	90% core, depleted blanket
Irradiation level	3,000 Mwd/ton	3,000 Mwd/ton (equivalent)
Irradiation time	50 days	250 days for blanket, 42 days for core
Uranium loading	10 tons	50 tons for blanket, 85 kg for core
Flux (thermal)	7.5×10^{13} n/cm²/sec	1.5×10^{14} n/cm²/sec for core, 1.5×10^{13} n/cm²/sec for blanket
Specific power	60 Mw/ton	6 Mw/kg U^{235}, 1.8 Mw/ton depleted U
Decay time	123 days	150 days for core, 107 days for blanket

ratio will also have a bearing on the effect of alloy composition.

Decay Time Required

The type of reactor, the method of irradiation, method of fuel recycle and refabrication techniques influence the amount of decay time required and thus the cost. These variables will be illustrated by the U^{237} problem associated with natural or enriched uranium.

Reactor type. The concentration of U^{237} (starting with $N_{26} = 0$) may be estimated by the equation

$$N_{27} = 8.8 \times 10^{6} P\phi [\lambda_{27} t - (1 - e^{-\lambda_{27} t})]$$

This equation was obtained by combining the relations

$$P = \phi \frac{WAE}{235} \frac{\sigma_f(25)}{3.38 \times 10^{16}}$$

and

$$N_{27} = \phi^2 \frac{WAE}{235} \frac{\sigma_c(25)\,\sigma_c(26)}{\lambda_{27}^2} [\lambda_{27} t - (1 - e^{-\lambda_{27} t})]$$

where P = specific power (Mw/kg U)
ϕ = neutron flux (n/cm²/sec)
t = irradiation time (sec)
λ_{27} = decay constant of U^{237} (1.19×10^{-6} sec^{-1})
N_{26}, N_{27} = U^{236}, U^{237} concentration (atoms/kg U)
W = wt of uranium (gm)
A = Avogadro number = 6.023×10^{23}
E = fuel enrichment (wt% U^{235})
$\sigma_f(25)$, $\sigma_c(25)$, $\sigma_c(26)$ = respective fission or capture cross sections; and 3.38×10^{16} = fissions/Mw/sec

For long irradiations, the U^{237} activity after the first irradiation cycle is directly proportional to the product of flux, specific power and irradiation time. Fuel material irradiated at a flux of 1.5×10^{13} will require 15 days' less decay time than that irradiated at 7.5×10^{13} for the same time and at the same specific power. In addition, for each factor of 2 that a fuel is lower in enrichment (for the same flux and specific power) 6.7 days' less decay time will be required for U^{237}.

About 30 days' less decay time is therefore required for a fertile blanket (0.3% U^{235}, $\phi = 1.5 \times 10^{13}$ n/cm²/sec) than for an internal-converter fuel (2% U^{235}, $\phi = 7.5 \times 10^{13}$ n/cm²/sec) irradiated to the same level. The decay time required for the highly enriched core is considerably longer than that for either the blanket or the internal converter, but less material will be stored (see comparison in Table 2).

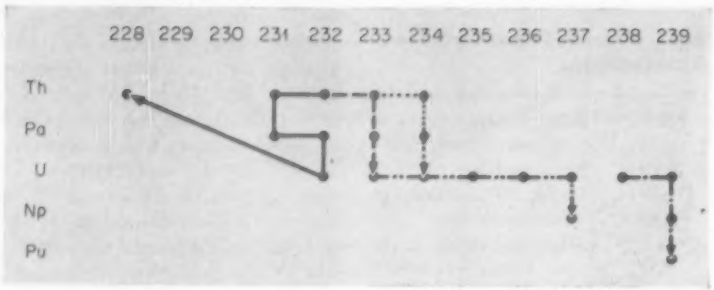

FIG. 2. Transmutation-isotope production routes in uranium and thorium fuels. Intermediate isotopes produced during irradiation lead to contamination and radiation problems. In recovering fissionable material, processor must decide if it is more economic to permit decay of intermediates or to separate them out

Recycle of a Pu^{239}-U^{238} fuel, such as that used in a fast breeder reactor, can be accomplished after only 20–30 days for neptunium decay. The Pu^{239} is separated from the U^{238} and fission products and then blended with fresh U^{238} before refabrication. If the Purex process is used, the separated U^{238}, as $2M$ uranyl nitrate, would be held back for 120 days for U^{237} decay. The tank charges for holdup of this U^{238} fraction would be approximately $120 per ton of U^{238} processed, with the additional inventory charges being approximately $40 per ton of U^{238} processed. Thus it is possible to minimize both the charge due to plutonium holdup and the increased cost of remote or semiremote refabrication with only a slight increase in the storage and inventory costs for U^{238} holdup.

Method of irradiation. The decay time of reactor-discharged material depends upon the method of irradiation, whether continuous (homogeneous type of operation) or batch, and, if batch, upon the number of irradiation cycles.

For any batch cycle, before (effectively) an infinite number of cycles has been accumulated, the concentration of transmutation isotopes of fertile and fissionable materials is lower than that for the continuous irradiation at steady state. For an infinite number of batch irradiations, the transmutation-product activities are the same as for the continuous case.

Fission-product activities are always lower in continuously discharged than in batch-discharged material.

The principal transmutation-product activity in U^{235} or natural uranium is U^{237}. The difference between the U^{237} produced during a batch and a continuous irradiation depends on the U^{236} content. In batch irradiation the U^{236} content depends on both the degree of its removal in a diffusion plant and the number of times the material has been recycled. If no U^{236} removal occurs, as in a diffusion plant, the U^{237} activity after any cycle of batch irradiation is $N_{26}/N_{25} \times \sigma_c(26)/\sigma_c(25)$ times the U^{237} activity resulting from continuous irradiation. For an infinite number of batch cycles this ratio is 1/1, but, for example, after 16 cycles at a flux-time of 10^{21} n/cm² per cycle, the ratio is $\sigma_c(26)/\sigma_c(25)$, since N_{26}/N_{25} is 1/1.

Method of recycle. The effect of recycle of both fertile and fissionable material through a diffusion plant, such as would be the case in a one-region slightly enriched reactor, can be illustrated by the following analysis.

After infinite recycle of slightly enriched uranium with the burned U^{235} replaced in each pass, the U^{236}/U^{235} atom ratio charged to the reactor will be approximately 7.5/1. However, if the reprocessed fuel is recycled through the gaseous-diffusion plant after each reprocessing cycle with 25% removal of U^{236} per pass, then this ratio is only 0.265/1 at a ϕt of 10^{21} n/cm² and 0.636/1 at a ϕt of 4×10^{21} n/cm². The effect of the decreased U^{236}/U^{235} ratio is illustrated in Fig. 1.

Refabrication techniques. The permissible activity levels in natural or enriched uranium are set by the U^{237} content. For direct refabrication, the activity of U^{237} must have decayed to 1×10^{9} dpm/kg U, which is only 3 mr/hr at direct contact, whereas semiremote fabrication may be performed if the activity is less than

4×10^{12} dpm/kg U. The decay time may therefore be shortened by 80 days if semiremote fabrication is used. Since 150 days would be needed for enriched uranium to decay to 1×10^9 dpm/kg (corresponding to $\phi t = 10^{21}$ n/cm², $\phi = 10^{14}$, 16 cycles), semiremote fabrication can save over 50% of the inventory and storage charges.

To re-enrich uranium in a diffusion plant, inventory charges will accrue for the period necessary for U^{237} to decay to background. Alternatively, the burnup can be replaced with fully enriched uranium and semiremote fabrication used after a shorter cooling period (perhaps half as long). Since the value per unit of contained U^{235} increases with enrichment, it is necessary to balance the increased U^{235} and semiremote fabrication costs of the second scheme against the higher inventory charges of the first.

Two Hypothetical Cases

Table 2, listing the characteristics of two hypothetical reactors, gives some indication of the enrichment, irradiation and decay problems associated with reactor fuels.

The two-region reactor offers a somewhat different situation with regard to fuel inventory and storage costs since the inventory charge for decay of the blanket is negligible. If the blanket-decay inventory is neglected, it is necessary to balance the core-decay charges against the slightly enriched fuel to compare reactors. To have a core-decay inventory charge equivalent to that for the single-region reactor, it is necessary to have 31% burnup of U^{235} with the full 150-day decay time. The decay time and U^{237} activity associated with various burnups to maintain an annual inventory charge for the core equal to that for the one-region reactor are indicated in Table 3.

Charges due to plutonium holdup may be reduced by processing after 2–3 weeks' decay and holding the uranium product for U^{237} decay. The uranium product may be stored as uranyl nitrate solution containing 400 gm of uranium per liter.

Thorium and U^{233}

Thorium decay brings up several additional problems. The activity of Th^{234} takes on the order of a year to decay to 10^9 dpm/kg Th; and Th^{228}, which builds up during the pre-Thorex decay time, decays with a 1.9-yr half-life.

However, the activity of infinitely recycled thorium may be kept to less than five times that of natural thorium by stringent limitations on reactor flux, irradiation level and decay time. The irradiation level must be below 4,000 gm/ton* and the reactor flux above 3×10^{13} n/cm²/sec to meet these specifications with reasonable decay times (>10 days) (*3*). Higher irradiation levels will require semiremote refabrication of the thorium products.

The specifications for semiremote handling based on 30 mr/hr at 1 ft will allow the total radioactivity to grow to 20 times that of natural thorium. This will increase the allowable irradiation level to 8,000 gm/ton at 3×10^{13} n/cm²/sec flux and zero decay time. The maximum permissible pre-Thorex decay time at 6×10^{13} n/cm²/sec flux and 8,000 gm/ton would be 90 days. Assuming that the Th^{228} chain contributes 75% of the total allowable activity, semiremote handling will also allow 150 days' less decay time for the Th^{234} decay. Thus, the thorium inventory charges may be decreased by a factor of 2 by using semiremote refabrication methods.

The problem with U^{233} is different since the pre-Thorex decay time has no effect upon the activity. The only decay time required is the total time for Pa^{233} decay, which may be held back in waste catch tanks. The activity of Th^{228} and its daughters in stored U^{233} builds up with a half-life of 1.9 yr and reaches a maximum in 10 yr. Thus U^{233} products will have to be repurified remotely. A kilogram of U^{233} metal (spherical) recovered from 4,000 gm/ton material would read 300 mr/hr at 1 ft after 3.5 months' storage. This activity level will allow only 4 hr/week working time when using the limits of 1 ft distance for hands and 2 ft for total body. To maintain a 40-hr/week working time for semiremote fabrication, the distance from source to hands must be set and maintained at 3.5 ft and the total body at 5½ ft. Twelve hours' semiremote operation working time per week may be allowed if the U^{233} product has aged only 30 days since repurification and fabrication is performed at a distance of 1 ft.

The method of irradiation has no effect on the activity of Th^{234} in thorium irradiation if it is assumed that thorium will not be returned to a reactor until after the Th^{234} activity has decayed at least 100-fold.

TABLE 3—Burnup Effect on U^{237} in Two-Region Reactor*

Burnup of U^{235} (%)	*Decay time necessary to limit core inventory cost to same as that for one-region reactor*	*U^{237} activity (dpm/kg U^{235})*	*Radiation level at 1 ft from U^{237} from 1 kg U^{235} (mr/hr)*
5	20	2.5×10^{14}	2,300
10	44	3.6×10^{12}	330
20	91	4.0×10^{11}	3.7
30	139	3.1×10^{9}	0.03
31	150	1×10^{9}	0.01
40–80	150		

* Basis: ~90% U^{235}; inventory cost based on value of fuel after burnup. The surface activity of a metal sphere weighing 1 kg with U^{237} activity of 3.6×10^{12} dpm/kg = 110 mr/hr. Solutions of U^{235} containing 1 kg of U^{235} at 450 gm U^{235} per liter would have surface activities 100 times that given for pure metal (based on standard containers). Fission-product activity with 90% removal = 120–180 watts/kg U^{235} = 5–8 $\times 10^{16}$ dpm/kg U^{235}.

* The term gm/ton for thorium irradiation corresponds to grams of mass 233 per metric ton of thorium.

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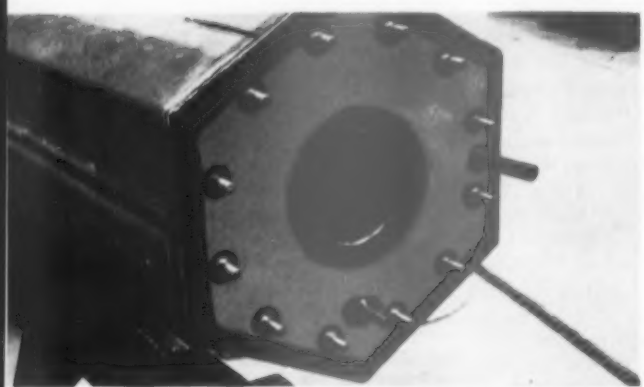
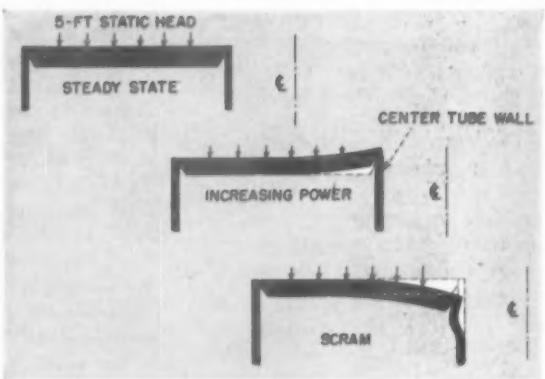
1 INSTALLING MODERATOR CANS (those with open center tubes) and reflector cans in 11-ft dia core tank. Loading pattern is triangular; 31 fuel rods locate in center-tube channels. Extra channels provide flexibility in loading. Vent tubes extend above coolant level to permit escape of gases formed in graphite when irradiated. Core of SRE consists of 119 cans containing graphite; 57 of them zirconium moderator cans with center tubes. Of 62 reflector cans, 29 are identical to moderator cans except no center tubes. Outer reflector region consists of 33 stainless-steel cans ranging in size from full to half hexagon

Developing Zirconium Cans for SRE Moderator Graphite

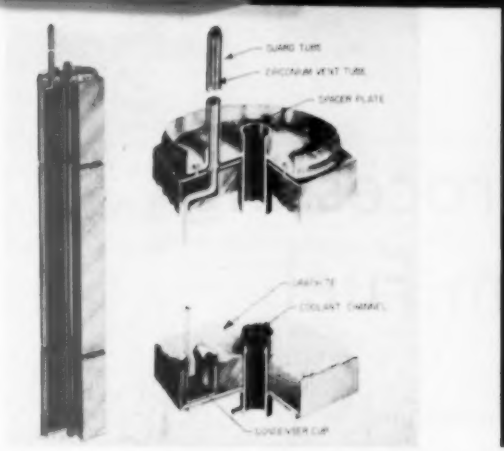
By JAMES A. LEPPARD
Atomics International, Canoga Park, California



6 ONE OF FIRST CANS to be completed. Heads were cut from 100-mil plate. Large center tube was brake-formed from 35-mil flat stock, welded, swaged, and sized. Small tubes, upper right for evacuation and lower right for vent, were formed from flat stock on Yoder mill

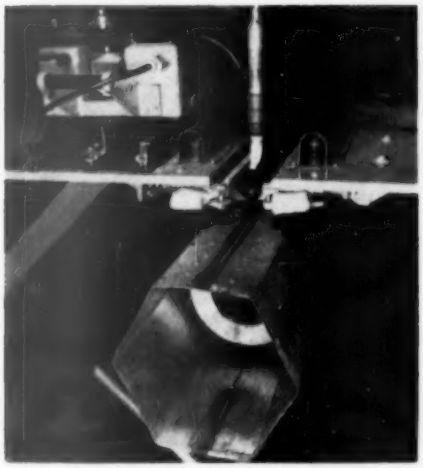


7 PRINCIPAL PROBLEM encountered during development of first cans was thermal stress just after scram. As power is increased, center tube expands more rapidly than head and shell, and when reactor is scrammed, center tube contracts more rapidly because of faster coolant flow



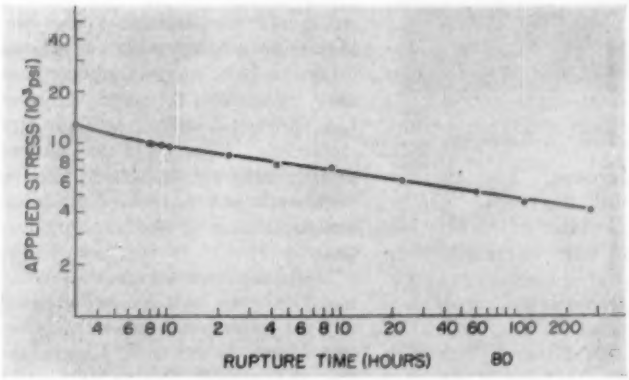
2 MODERATOR CAN is 10.8 in. across flats and 10 ft high with 2.8-in. dia (i.d.) center tube. Moderator is canned because absorption of sodium coolant by porous graphite would decrease moderating ability and structural integrity

3 HEXAGONAL OUTER SHELL of can is made up of six panels formed from 35-mil unalloyed zirconium sheet by profile dies on power brake at room temperature. Panels are stress-relieved at 1,040° F for 15 min after forming, and then are vapor blasted and dry-machined to tolerances. Scalloped corners form, with adjacent cans, channels for control, safety and neutron source rods



4 WELDING MACHINE AND FIXTURE used to join panels is completely automatic. Inert-gas-shielded butt-welds were made at 4 in./min without filler material using $\frac{3}{32}$-in. dia tungsten electrode

5 ASSEMBLING GRAPHITE STRINGERS prior to pinning and fitting into can. Center tube and heads were manually welded in place using inert-gas-shielded tungsten arc. Despite considerable difference between thickness of panels and heads, joints were made without filler material. All welds were dye-checked and X-rayed



8 STRESS-TO-RUPTURE TEST of hafnium-free zirconium for various loadings at 1,000° F. Longitudinal grain, 35-mil sheets were tested in helium atmosphere. Note that at 200 hr the sustained load is only 4,300 psi; hence tube-to-head joint repeatedly failed in can-head flexure tests

9 PROBLEM WAS SOLVED when stiffening collar of 100-mil stock 1 in. long was substituted at top and bottom of center tube. Additional flexure tests were without failure through 3,000 cycles

Fused-Fluoride Processing of Spent Uranium Fuel

Fused fluorides can remove all but trace of plutonium and 95–100% of fission products (except Ru) from molten uranium. Decontamination is less than that obtainable with aqueous processing but would be sufficient for fast breeders

By ARCHIE G. BUYERS
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PYROPROCESSING OF IRRADIATED URANIUM FUELS (*1*) for removing plutonium and fission products includes such processes as molten-salt–molten-uranium fuel equilibration, refractory-oxide–molten-metal contacting, liquid-metal–uranium-fuel studies, and direct distillation of contaminants from melted fuels.

The molten-salt–molten-uranium equilibration process, particularly a UF_4 scrub, may be most useful for fast breeder reactors, i.e., those with large plutonium concentrations in the core and blanket. The fluoride process is suitable for use on a continuous basis to limit the core plutonium concentrations to that demanded for optimum reactor operation. Investigations of this system are reported here.

Uranium-metal fuel-rod fragments irradiated in the X-10 reactor at Oak Ridge to ~95 gm Pu per ton of U and cooled for 2½ yr were chosen for these studies. Fragments were melted at 1,573° K with UF_4, calcium and lithium fluorides and mixtures of these salts.

Experimental Procedure

The experimental steps include:

1. Degassing graphite crucibles.
2. Prefusing salts and salt mixtures.
3. Equilibrating salt and metal at 1,573° K.
4. Separating and dissolving phases.
5. Radiochemical analysis.

Degassing. The graphite crucibles were degassed by heating inductively to 1,573° K under a reduced helium atmosphere, evacuating the system to better than 1 micron, and repeating until no major pressure rise was noted upon reheating the crucible.

Prefusing. After cooling, the weighed salt, or salts, were introduced into the crucible, and the system was evacuated. Nearly an atmosphere of helium, purified by passing over uranium turnings heated to 873° K, was let into the system and the salt fused into a compact mass. The melt was placed in a fresh graphite crucible with a uranium fuel fragment previously cleaned with acid, dried and weighed.

Equilibration. The system was evacuated to 0.01–0.07 micron, and purified helium gas let in to nearly atmospheric pressure. The crucible contents were rapidly melted at 1,573 ±50° K and held at this temperature for ½ hr.

Phase separation. After the system had cooled, the crucible was removed from the inner quartz tube and sawed apart. Salt and metal were separated in a steel mortar and pestle.

Final salt traces were removed from the ingot surface with a 6*N*-HNO_3–saturated-H_3BO_3 mixture. This wash was added to the solution prepared from the separated salt that had been dissolved in a similar acid mixture (vacuum distillation of the final salt traces included in the metal ingot was less effective than the acid-mix washing). The remaining uranium metal was dissolved in acid.

The inner surfaces of the quartz fusion tube and crucible fragments were leached with hot HNO_3-H_3BO_3 followed by boiling water. These liquids were also added to the salt-phase solution.

All solutions were diluted or concentrated to as small a volume as consistent with solubility considerations.

Radiochemical analysis. A control solution made up from a cleaned unprocessed uranium fragment and solutions prepared from the separated melt phases were transported from the gloved box to another box designed specifically for plutonium analyses. Plutonium analyses were carried out using the LaF_3 coprecipitation procedure (*2*) modified to include only one LaF_3 precipitation and no hydroxide metathesis. Dilutions of the solutions directly prepared from melt components were removed to a radiochemical analytical laboratory for fission-product analysis (*3*).

Mechanical separation of the salt and metal using the steel mortar and pestle did not remove salt products frozen into fissures in the metal because the two phases melt at about the same temperature. In several experiments the outer portion of the uranium residue was dissolved for analyses separate from those carried out upon the inner

TABLE 1—Fused Fluoride and Uranium Fuel Equilibration Results at 1,573° K*

Experiment No.	Salt-mixture composition (mole fractions)			% Pu in separated phases			Pu activity ($\times 10^8$ α/sec)		N_{PuF_3}† / N_{Pu}	Thermodynamic values	
	UF_4	CaF_2	LiF	Salt	Outer U	Inner U	Salt	U		K_e‡	$-\Delta F°$§ reaction
1	1.000			86.7	7.5		12.84	1.11	84.8	4.4	4.6
2	1.000			87.2	5.3		16.64	1.02	137.8	5.2	5.1
3	1.000			91.3	4.4		18.00	0.87	151.4	5.3	5.2
4	1.000			94.4	4.2	5.3	24.21	1.71	81.8	4.3	4.6
5	1.000			77.5	14.3	7.2	23.70	2.21	67.2	4.1	4.5
6	0.17	0.22	0.61	94.7	2.2	0.2	15.83	0.07	66.2	6.3	5.3
7	0.46	0.14	0.40	81.7	15.5	1.1	16.19	0.22	91.4	5.5	5.7
8	0.02		0.98	88.2	7.6	1.0	16.68	0.19	14.4	6.5	5.8
9	0.02	0.98		71.8	29.1	7.1	8.75	0.87	3.1	3.8	4.2
10 Cycle 1	0.17	0.22	0.61	85.4			42.50	9.05	11.7	3.5	4.0
Cycle 2	0.17	0.21	0.62	16.7			8.33	0.72	31.3	4.9	5.0
Cycle 3	0.17	0.21	0.62	0.8	0.6	0.1	0.40	0.08	14.0	3.8	4.1
									Average	4.8	4.8

* Melting time was 30 min; melts were contained in graphite crucibles.
† Approximate mole-fraction ratio of Pu in the salt to Pu in analyzed metal (the analyzed metal was less than that introduced).
‡ Equilibrium constant for reaction: $\frac{1}{3}Pu + \frac{1}{4}UF_4 \rightleftharpoons \frac{1}{3}PuF_3 + \frac{1}{4}U$.
§ ΔF (reaction) = kcal/equiv. at 1,573° K for the reaction under ‡.

or clean-metal solution in an attempt to ensure removal of these last salt traces. This procedure was found effective, but required solution of at least one-third of the processed uranium ingot.

One three-cycle equilibration of molten uranium with a UF_4-CaF_2-LiF salt mixture was carried out. The procedure for this experiment was similar to that used in the foregoing investigation except that the mechanically separated uranium metal from the first cycle was washed with acid and placed in a new crucible with fresh salt and remelted. A similar operation constituted the third cycle.

Duplication of single contacting runs was carried out with cold natural uranium for equilibration of metal and the CaF_2-LiF mixture. Uranium analyses were run on the salt phase (*4*).

Results and Discussion

Preliminary tracer-activity studies had shown UF_4 to be an effective extractant for plutonium (*5*). There is some question, however, as to whether the tetra- or trivalent uranium fluoride is the active oxidant.

Plutonium-uranium equilibria. Both UF_4 and UF_3 can exist in a melt containing the tetrafluoride and uranium metal according to the equilibrium

$$\frac{1}{4}\,UF_4 + \frac{1}{12}\,U = \frac{1}{3}\,UF_3 \quad (1)$$

The trifluoride is reported to decompose above 1,273° K (*6*). More recently, UF_3 has been reported to disproportionate at all temperatures from 973° K, the rate of production of UF_4 increasing with temperature (*7*). Other investigators studying UF_3-bearing systems report the presence of large amounts of UF_4 (*7*). It is also reported that UF_3 contained in Inconel does not disproportionate below 1,523° K (*9*).

Thus the extraction of plutonium as

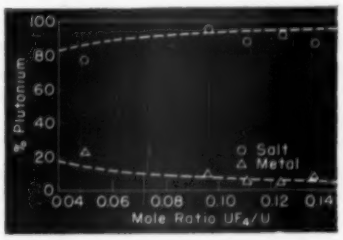

FIG. 1. Plutonium distribution between salt and metal as function of UF_4/U mole ratio

the trifluoride might be represented by one of the following equations

$$\frac{1}{3}Pu(l) + \frac{1}{4}UF_4(l) \rightleftharpoons \frac{1}{3}PuF_3(l) + \frac{1}{4}U(l) \quad (2)$$

$$\frac{1}{3}Pu(l) + \frac{1}{3}UF_3(l) \rightleftharpoons \frac{1}{3}PuF_3(l) + \frac{1}{3}U(l) \quad (3)$$

$$\frac{1}{5}UF_4(l) + \frac{1}{15}UF_3(l) + \frac{1}{3}Pu(l) \rightleftharpoons \frac{1}{3}PuF_3(l) + \frac{4}{15}U(l) \quad (4)$$

For Eqs. 2 and 3, if the uranium fluoride and uranium metal are assumed pure and therefore at unit activity, the equilibrium constant is

$$K_e = (a_{PuF_3}/a_{Pu})^{1/3} \quad (5)$$

where a_{PuF_3} and a_{Pu} are the thermodynamic activities of plutonium trifluoride and plutonium metal. The equilibrium constant for Eq. 4 is

$$K_e = \left(\frac{a_{PuF_3}}{a_{Pu}}\right)^{1/3} \frac{(a_U)^{4/15}}{(a_{UF_4})^{1/5}(a_{UF_3})^{1/15}} \quad (6)$$

The uranium metal-to-fluoride ratio should have a small effect upon the calculated equilibrium constant.

Equation 5 should in any case approximate the equilibrium constant. If we assume ideal solution wherein the thermodynamic activity equals solute concentration or mole fraction, the equilibrium constant is given by

$$K_e = (N_{PuF_3}/N_{Pu})^{1/3} \quad (7)$$

where N_{PuF_3} and N_{Pu} are mole fractions of PuF_3 in the salt phase and Pu in the uranium phase.

Equilibrium constants were calculated for all experiments in which a completely salt-free uranium ingot was obtained. Table 1 shows an average calculated K_e of 4.8. Pseudo mole fractions were obtained by dividing the Pu^{239} α-disintegration rate of the analyzed material by the total moles of the phase containing the Pu.

The standard free energies given in

TABLE 2—Free Energies of Formation of Fluoride Salts (at 1,500° K)

Salt	$-\Delta F^\circ$ form. (kcal/equiv.)	Reference
SrF_2	116	11
CaF_2	115	11
BaF_2	115	11
LaF_3	113	11
LiF	112	11
CeF_3	111	11
NdF_3	109	11
PrF_3	109	12
SmF_3	105	12
MgF_2	100	11
PuF_3	96	13
CsF	87	11
UF_4	88	11
UF_3	89	11
PuF_4	81	13

Table 1 for Eq. 2 were calculated from

$$\Delta F^\circ = -RT \ln K_e \qquad (8)$$

From these data it appears that the standard free energy of the reaction representing plutonium extraction is −5 kcal at about 1,573° K.

Also tabulated in Table 1 are the plutonium decontamination data expressed as 100 times the radioactivity measured in a given phase after equilibration divided by the activity introduced into the experiment. These data reveal that equilibration of UF_4 and irradiated uranium metal at 1,573° K for 30 min results in the movement of all but a negligible trace of plutonium into the fluoride phase.

Data correlation. To simplify data correlation and interpretation, it is assumed the active plutonium oxidant in the equilibration salt is UF_4.

A series of U-UF_4 equilibrations at 1,573° K was carried out to define a range of salt-to-U mole ratios that permit optimum plutonium extraction. Figure 1 shows that, as the mole ratio UF_4/U increases, the amount of plutonium in the salt phase rapidly approaches saturation. Obviously, further increases in this mole ratio were not advantageous.

The curves plotted in Fig. 1 were calculated from

$$\text{Mole ratio } \frac{UF_4}{U} = \left(\frac{1}{K_e}\right)^3 \frac{F_{PuF_3}}{1 - F_{PuF_3}} \qquad (9)$$

which is developed from the equilibrium constant for Eq. 2

$$K_e = \left(\frac{N_{PuF_3}}{N_{Pu}}\right)^{1/3} = \left(\frac{\dfrac{F_{PuF_3}C}{\text{moles } UF_4}}{\dfrac{F_U C}{\text{moles U}}}\right)^{1/3}$$

$$= \left(\frac{F_{PuF_3}/\text{moles } UF_4}{(1 - F_{PuF_3})/\text{moles U}}\right)^{1/3} \qquad (10)$$

where F_{PuF_3} is the fraction of plutonium originally present in the spent fuel that is found in the salt after equilibration and F_U is the fraction in the equilibrated uranium phase. C is a constant of proportionality. In these calculations the average experimental equilibrium constant, 4.8, was substituted for K_e.

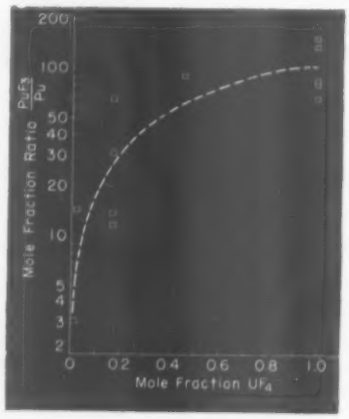

FIG. 2. Plutonium distribution as function of mole fraction UF_4 mixtures of calcium, lithium, and uranium fluorides

The study was extended to include other suitable fluorides. In choosing salts to mix with UF_4 for equilibration with uranium, an effort was made to select those fluorides that might themselves act as extractants by distillation of the metal that had been the fluoride cation. It is entirely probable that, during a salt-uranium fusion at 1,573° K, uranium reduces the salt cation to form equilibrium concentrations of uranium, plutonium, and fission-product fluorides. This can occur when the metals formed by reduction are volatile.

The more important inorganic fluorides are tabulated with their free energies of formation at 1,500° K in Table 2. The two most promising salts were lithium and calcium fluorides. The melting point of calcium fluoride was too high, but mixtures of the two salts were relatively low-melting. Accordingly mixtures of calcium, lithium and uranium fluorides were prefused and equilibrated with cleaned uranium at 1,573° K for ½ hr.

The distribution data are summarized in Table 1. In most instances inner and outer metal phases were separated by solution of the outer ingot surface in nitric acid to ensure complete salt removal. Table 1 indicates that regardless of fluoride or fluoride mix used, plutonium removal into the salt phase was appreciable, but not quite complete. In the three-cycle experiment, the plutonium remaining in the clean uranium residue after the third equilibration with salt was negligible.

In Fig. 2, experimental plutonium-mole-fraction ratios from Table 1 are plotted as a function of the mole fraction of UF_4 in the salt scrub. The broken curve was calculated from the average equilibrium constant 4.8 and the equation

$$(N_{PuF_3}/N_{Pu})^{1/3} = K_e(N_{UF_4})^{1/3} \qquad (11)$$

The thermodynamic activity of uranium metal is taken as unity.

In cases where no UF_4 was added to the salt mixture, small amounts of uranium and plutonium fluorides were formed either by distilling calcium or lithium metal after reduction from the salt phases or from impurities in the prepared salts. Uranium analyses

TABLE 3—Fission-Product Distribution at 1,573° K with Fluoride-U Equilibration

Composition		Percent of fission product in each phase											
		Ce		Rare earths		Sr		Cs		Te		Ru	
Salt	Mole frac.	Salt	Met.	Salt	Met.	Salt	Met.	Salt	Met.	Salt	Met.	Salt	Met.
UF_4	1.00	52.4	26.6	52.5	29.5	49.8	32.7	37.2	15.8	10.2	73.1	3.2	96.0
UF_4	Trace	37.6	45.8	32.7	55.6	23.5	58.3	12.4	47.7	30.1	35.4	6.0	81.5
CaF_2	0.26												
LiF	0.74												

were made upon duplicate cold experiments to determine the UF_4 concentration. Plutonium-extraction data obtained in these instances were not so favorable to decontamination as with UF_4 in the melt.

The data for the three-cycle experiment, although shown in Fig. 2, are not included in the curve interpretation since this experimental procedure differed from the one-cycle investigation in that there was no metal-phase analysis until the completion of the third cycle. The PuF_3/Pu-mole-fraction ratios appearing in Table 1 and Fig. 2 were calculated by assuming that plutonium not in the salt was present in the uranium residue and in the subsequent cycle. This was not necessarily true since the processed metal was not thoroughly cleaned of salt between cycles inasmuch as it was desired to lose a minimum amount of uranium.

The theoretical plutonium extraction by a fluoride mixture containing UF_4 is considered important from an engineering standpoint. The curves plotted in Fig. 3 were calculated from the average experimental equilibrium constant, 4.8, and the expression

$$\frac{F_{PuF_3}}{1 - F_{PuF_3}} = (K_e)^3 (N_{UF_4})^{3/4} \frac{\text{moles salt}}{\text{moles U}} \qquad (12)$$

which is based on Eq. 2 and the equilibrium constant

$$K_e = \left(\frac{F_{PuF_3}}{\dfrac{\text{moles salt}}{F_{Pu}}} \right)^{1/3} \left(\frac{N_U}{N_{UF_4}} \right)^{1/4} \qquad (13)$$

where N_U approximated unity, because

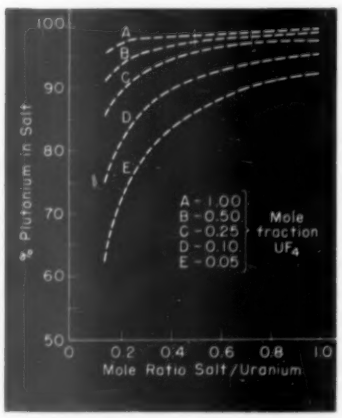

FIG. 3. Plutonium extraction as function of mole fraction UF_4 and fluoride/metal mole ratio

the Pu and fission-product concentrations were very small.

It is possible to predict from Fig. 3 the plutonium extraction by a given melt of known mole fraction UF_4.

Fission-product distribution. The fission-product distribution was measured by radiochemical analysis (*3*) of diluted uranium and salt solutions prepared from phases separated after equilibration. Data obtained from experiments in which the melt components were separated mechanically and the ingot outer surface was not removed and separately analyzed are summarized in Table 3.

Not all the experiments tabulated and discussed in the foregoing plutonium section were evaluated for fission-product distribution. It was decided that percentages determined by three UF_4 experiments were representative.

Distribution values reported for salt phases are low and those tabulated for the metal phases are high as a result of salt traces that were included in or attached to the outer uranium ingot surfaces. Despite this fact, the data show reasonably effective uranium decontamination of cerium, rare earths, strontium and cesium. Tellurium and ruthenium do not move appreciably into the salt scrub.

Examination of CaF_2-LiF equilibration data reveal less effective cerium, rare-earth, strontium and cesium decontamination and again little or no ruthenium present in the salt. Tellurium decontamination is apparently greater than that achieved with UF_4. As the tellurium removal probably was the result of distillation of the metal, variation of these values is not unexpected. Interpretation of these data requires cognizance of the same factor that existed in the UF_4 data, i.e., the existence of salt traces in the surface of the outer uranium residue. The effect of these traces probably is more pronounced on the present data as the melting points of the CaF_2-LiF mixtures were much closer than the melting point of UF_4 to that of uranium metal.

Uranium-fused fluoride equilibrations at 1,573° K in which at least the outer one-third of the processed uranium ingot was dissolved in acid and analyzed separately from both salt and metal solutions are summarized in Table 4. This table shows that fission-product distribution is much more favorable to decontamination than previous experiments indicated.

TABLE 4—Fission-Product Distribution in Fused Fluorides and U*

Expt. No.	Phase	% activity measured: Ce	Rare earths	Sr	Cs
6	Salt	52.8	47.2	21.9	23.3
	U_{outer}	32.7	39.9	24.3	22.7
	U_{inner}	4.5	4.1	2.9	3.3
7	Salt	39.3	36.3	31.7	34.5
	U_{outer}	50.4	51.3	23.3	26.0
	U_{inner}	0.9	0.4	0.3	0.4
8	Salt	78.8	86.9	85.7	28.9
	U_{outer}	8.3	6.2	5.1	0.9
	U_{inner}	0.1	0.2	0.1	Trace
9	Salt	51.6	86.1	78.0	67.4
	U_{outer}	4.6	6.1	5.1	2.2
	U_{inner}	0.1	0.2	Trace	Trace
10 Cycle 1	Salt	51.6	53.0	54.7	27.9
Cycle 2	Salt	29.8	31.0	31.1	6.9
Cycle 3	Salt	0.8	0.9	0.8	15.1
	U_{outer}	0.6	0.7	0.5	0.3
	U_{inner}	0.2	0.2	Trace	0.4

* After equilibration at 1,573° K, salt and metal were mechanically separated, outer portion of uranium ingot dissolved in acid, and fission products measured in salt, outer U and inner U.

It is not possible to correlate the fission-product extraction data by means of equilibrium constants. Apparently the direct equilibrium, such as

$$\tfrac{1}{3}Ce + \tfrac{1}{4}UF_4 = \tfrac{1}{3}CeF_3 + \tfrac{1}{4}U \qquad (14)$$

was disturbed by some competing reaction. Very probably oxide in the system caused the disturbance. Plutonium correlation is possible because oxide has little effect on the plutonium distribution (*10*).

* * *

The author expresses his appreciation to D. Cubiciotti for the intelligent guidance that he offered in carrying out these studies. The work reported here is given in more detail in reference 14.

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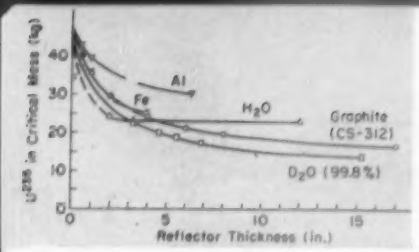

FIG. 1. Reflectors reduce critical mass of Oy(93.5) spheres. Graph shows amount of U^{235} in a critical mass of density 18.8 gm/cm³ as function of reflector thickness

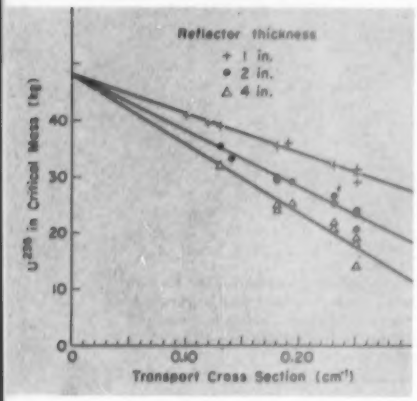

FIG. 2. Critical mass decreases with reflector macroscopic transport cross section. Data are in table

Critical Masses of Oralloy Assemblies

Nuclear-safety data from critical experiments tell the effects of reflection, shape, density, concentration, and dilution on enriched-U configuration. One can also draw some conclusions concerning Pu and U^{233}

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CRITICAL-MASS information such as that presented recently by Callihan (*1*) provides the data needed for realistic answers to questions of nuclear safety under many conditions. Where immediate surroundings are readily controlled, it is straightforward to apply safety factors to these basic data. Thus one can arrive at rules that can be trusted to prevent criticality accidents if the rules are properly implemented.

In the general case, where a great variety of techniques may be represented, it is frequently difficult to define a completely appropriate set of conditions. Over-all regulations are likely to become so cumbersome as to be easily misapplied. An operation deemed safe under established rules may involve large nuclear risk among the consequences of an unconsidered, nonnuclear accident. On the other hand, blind observance of rules may lead to unnecessary expense especially in the design of large processing plants. A proper understanding of criticality data will alleviate the difficulties of situations like these.

The following material is a summary of data on critical configurations of fissionable metals. It is presented in a form that emphasizes the influence of conditions commonly of concern in nuclear-safety questions. All of the data are empirical. Each point on the curves and each entry in the tables represent one or more measurements. In some cases a little theory has been used on the data before presentation. The most significant example is the deduction of effective transport cross sections from a tremendous number of measurements on reactivity contributions at various radii in critical assemblies.

The bulk of our data are derived from experiments with oralloy-93.5 [Oy(93.5)]. The term refers to uranium metal in which the U^{235} is enhanced so that it constitutes 93.5% of the material.* There are also enough data for plutonium and U^{233} to establish some general relations about their criticality relative to uranium.

The specific information that we present includes the influences exerted on critical mass by various reflectors, shape of fissionable material, variations in U^{235} concentration, variations in material density, and dilution by graphite and hydrogenous materials.

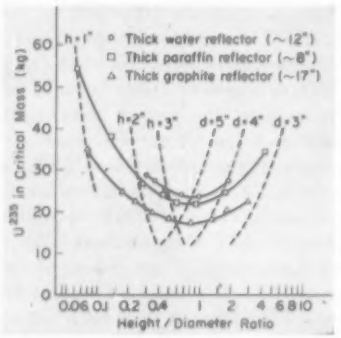

FIG. 3. Changes of shape have small effect on critical mass. Amount of U^{235} in critical-mass cylinder of Oy(93.5) is shown as function of ratio of height to diameter for several reflectors

Reflectors

Reflectors are of general concern in nuclear safety. They can be introduced either intentionally, to increase the criticality or improve the uranium economy of a reactor, or incidentally, as they occur in the production of metal and the processing of fuels. Examples belonging to one or both categories are graphite, beryllium, light and heavy water, oxide refractories, molten salts, and steel.

*The term "oralloy" was coined about 1949 to dispel confusion. Oralloy was offered as a name for enriched material and U^{235} and U^{238} were reserved for the isotopes themselves.

Dependence of critical mass on reflector thickness is illustrated in Fig. 1. The striking difference between the shapes of the curves for H_2O and D_2O reflects, of course, the small capture cross section of deuterium relative to hydrogen. In general, the critical mass of a bare isolated sphere of Oy(93.5) metal, 48 kg U^{235}, would be reduced to 19–23 kg by a thick reflector of nearly any common material, such as Al, Fe, Cu, H_2O, etc. Reflectors known to be exceptionally effective are Be, D_2O, graphite in large thicknesses, U, W, and WC.

Where direct information is unavailable, the relative effectiveness of a neutron reflector can be judged by comparing its macroscopic transport cross section with those of materials of which the reflection efficiencies are known. The table lists critical masses for standard thicknesses of various reflectors along with the transport cross sections as deduced from measurements of material effectiveness in critical assemblies. For modest thicknesses, the critical mass decreases nearly linearly with increasing macroscopic transport cross section. This is shown in Fig. 2, which presents the data of the table below graphically.

Shape

The critical mass of enriched uranium metal is insensitive to minor changes of shape. For example, it can be seen in Fig. 3 that the critical mass of an Oy cylinder with a thick graphite, water, or paraffin reflector changes only 20% as the ratio of height to diameter ranges from ½ to 2. As reflector effectiveness is decreased, critical mass becomes somewhat more dependent on shape, as shown by Fig. 4.

Limiting critical dimensions of Oy(93.5) metal cylinders have been obtained by R. E. Gwin at Oak Ridge from guided extrapolations of the data in Figs. 3 and 4. The results are

Reflector	*Infinite-rod diameter*	*Infinite-plate thickness*
None	4.9 in.	2.4 in.
Thick water	2.9	0.75
Thick graphite or U	2.3	0.5

These dimensions are considered principally suitable for orientation, and are less useful than the corresponding ones for solutions (*1*), which have direct applicability to processing plants.

U^{235} Concentration

The critical mass of oralloy metal as a function of average U^{235} concentration has been determined for nearly spherical oralloy in a thick reflector of natural uranium and for a series of unreflected oralloy cylinders. The latter

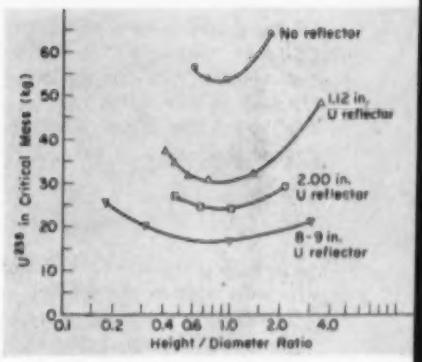

FIG. 4. Sensitivity of critical mass to shape increases with decrease of reflector thickness as shown by these data on critical-mass cylinders of Oy(93.5) in natural U reflectors

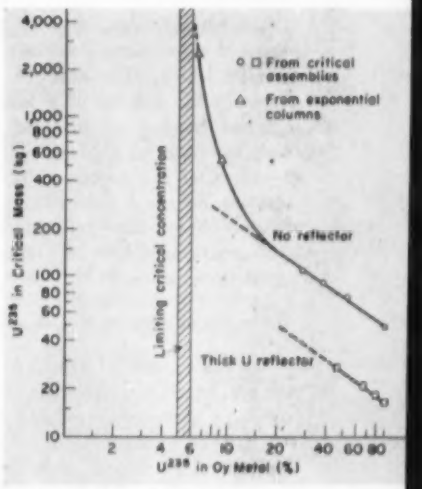

FIG. 5. These data show concentrations of U^{235} in oralloy required to make spheres critical

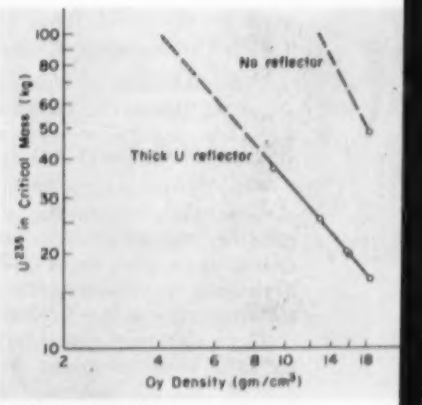

FIG. 6. Critical mass is proportional to Oy density$^{-1.5}$ when Oy(93.5) spheres are reflected by normal-density natural uranium. Relation computed for bare spheres is shown as broken line

Critical Masses of Oy(93.5) Spheres with Various Reflectors*

Reflector	*Density* (gm/cm^3)	*Reflector thickness* 1 in.	2 in.	4 in.	∞	*Effective* Σ_{tr} (cm^{-1})
Be (QMV)	1.84	29.2	20.8	14.1	—	~0.25
BeO	2.69	—	21.3	15.5	~8.9	—
WC	14.7	—	21.3	16.5	~16.0	—
U	19.0	30.8	23.5	18.4	16.1	0.25
W-alloy (~92% W)	17.4	31.2	24.1	19.4	—	~0.25
Paraffin	0.91	32.6	—	—	21.8	—
H_2O	—	—	~24.0	22.9	22.8	—
D_2O	—	—	(27)	21.0	~13.6	—
Cu	8.88	32.4	25.4	20.7	—	0.23
Ni	8.88	33.0	25.7	(21.5)	~19.6	0.23
Al_2O_3	2.76	35.1	—	—	—	—
Graphite (CS-312)	1.69	35.5	29.5	24.2	~16.7	0.18
Fe	7.87	36.0	29.3	25.3	23.2	0.19
Zn	7.04	—	29.8	25.0	—	0.18
Th	11.48	—	33.3	—	—	~0.14
Al (28)	2.70	39.3	(35.5)	(32)	<30.0	0.13
Ti	4.50	39.7	—	—	—	0.12
Mg	1.77	41.0	—	—	—	0.10

* Data are adjusted to the standard reflector thicknesses shown. Density of Oy(93.5) is 18.8 gm/cm^3. Parentheses indicate interpolated data.

data, in conjunction with Figs. 3 and 4 or by conversion through buckling calculations, can yield the critical mass of bare spheres. When presented to apply to spheres, both sets of data, for concentrations >20%, are of the form

$$U^{235} \text{ critical mass} \propto (U^{235} \text{ concentration})^{-0.7}$$

Figure 5 shows the two relations and includes data in the very-low-concentration range that come indirectly from exponential-column measurements. The shaded region represents the estimated uncertainty in the limiting U^{235} concentration, below which an infinite mass of oralloy cannot be critical.

Oralloy Density

The radius of a critical system is proportional to the mean free path for nuclear processes. For example, if one decreases the density of a fissionable material causing an increase in the mean free path, he will have to increase the physical dimensions in the same ratio to maintain interaction probabilities. For a bare core and a fixed concentration of U^{235} in oralloy, this yields the relation

$$U^{235} \text{ critical mass} \propto (\text{Oy density})^{-2.0}$$

Critical masses of Oy(93.5) metal spheres for various average densities of oralloy have been measured in thick reflectors of normal uranium at normal density. The diluent was air. The resulting relation, Fig. 6, is of the form

$$U^{235} \text{ critical mass} \propto (\text{Oy density})^{-1.2}$$

For oralloy in thick normal uranium, where the density of either can vary, it follows that

$$U^{235} \text{ critical mass} \propto (\text{Oy density})^{-1.2} (\text{U density})^{-0.8}$$

Dilution of Oralloy

Water is the oralloy diluent most likely to be of concern in nuclear-safety questions, though many common hydrogenous materials can be expected to have similar nuclear properties.

Water. Criticality measurements made on uniformly spaced lattices of Oy(93.5) cubes immersed in water showed that for each cube size the critical mass occurs at a face-to-face spacing of about ⅝ in. between oralloy units. The dependence of the critical mass of Oy(93.5)-water mixtures on the size of oralloy units is shown in Fig. 7.

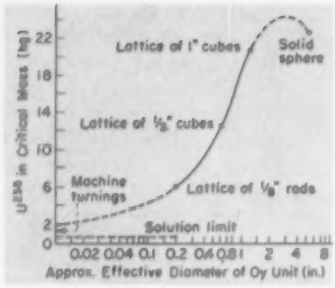

FIG. 7. Variation of critical mass of water-flooded lattices of Oy(93.5) rods or cubes with unit size; all data at optimum spacings

(The dashed portion of that curve implies lack of detailed knowledge; one knows that H_2O in small units is less effective than a similar quantity of Oy(93.5) in the solid sphere, and therefore the effect of small dilutions must be to increase the critical mass.) The results in Fig. 7 underline the need for precautions against flooding accumulations of oralloy machine turnings, loose coils of wire, or similar pseudolattices of thin oralloy with either water or oil.

Graphite. The case of Oy(93.5)-graphite mixtures can be of interest as a prototype of a multiple casting in a graphite mold. Figure 8 gives critical masses of various stacks of 10½-in.-diameter oralloy metal plates separated by graphite and surrounded by a 2-in. graphite reflector. It is seen that for Oy plates in 0.315-in., 0.630-in., and 0.945-in. thicknesses the moderating effect of the graphite is outweighed by the decrease in oralloy density when one introduces graphite separators. Naturally this does not remain true for much thinner oralloy, since one would expect a behavior at large graphite dilutions somewhat similar to that shown for water in Fig. 7.

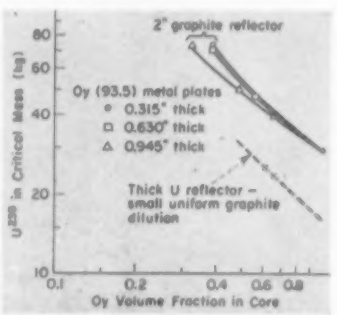

FIG. 8. Critical mass increases as Oy plates of 10½-in. diameter are separated by graphite disks. Data obtained from cylinders have been reduced to spherical equivalents

The dashed line of Fig. 8 is deduced from measurements of the relative effectiveness of graphite and oralloy at various positions inside a fast-neutron critical assembly surrounded with a thick uranium reflector. This curve, in the strict sense, is valid only for predicting the critical mass resulting from a small dilution by dispersed graphite. Data for similar computations exist for a large number of elements (*2*).

Pu^{239} and U^{233}

Critical masses of plutonium metal (at a density of 15.6 gm/cm³) and Oy(93.5) have been compared for bare and also reflected spheres. In all these cases the plutonium critical mass is roughly ⅓ that of oralloy. Not only is this ratio confirmed by effectiveness measurements in fast critical assemblies, but it is further deduced that the critical masses of plutonium and U^{233} (at a density of 18.5 gm/cm³) will be nearly identical.

Additional comments about unmoderated Pu and U^{233} systems are

- The relation, critical mass ∝ (density)$^{-2}$, stated before for bare oralloy, applies also to bare Pu or U^{233} metal.
- The ratio of the critical mass of a plutonium metal cylinder (height/diameter = 0.4) in a thick uranium reflector to that of a sphere in the same reflector was observed to be identical to the corresponding ratio for Oy(93.5).
- Pu^{240} present in Pu^{239} metal to less than ~10% increases the critical mass slightly. Measurements in fast critical assemblies indicate that the net fission production per atom of Pu^{240} is about ⅔ that of Pu^{239}.

For lack of more detailed information, then, critical masses of plutonium (ρ = 15.6 gm/cm³) and U^{233} metal are estimated by scaling U^{235} values downward by a factor of 0.35. In nuclear-safety considerations, the resulting uncertainties should be covered by generous safety margins.

* * *

All data presented here have been previously reported by the authors in LA-1958, which is classified, and LA-1958 (Deleted), which is not.

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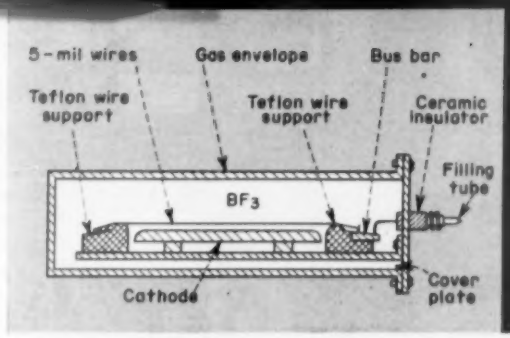

FIG. 1. Neutron-detecting spark counter

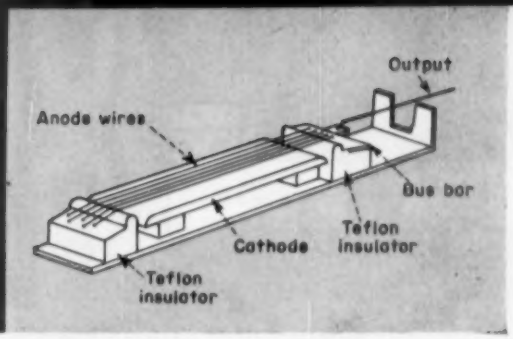

FIG. 2. Multiple-wire anode for spark counter

BF_3 Neutron Spark Counter

A BF_3-filled spark counter is sensitive to neutrons yet insensitive to gammas. Cadmium shielding plus paraffin moderation make it useful for fast-neutrons too

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WITH A ROSENBLUM spark counter (*1*) operated in a BF_3 atmosphere we have achieved a slow-neutron detection efficiency of 0.6%, and the system is completely insensitive to Co^{60} radiation at 500 r/hr. External circuitry required is limited to a high-voltage supply and a scaler.

Sensitivity

Detection efficiency is proportional to the probability of slow-neutron capture in B^{10} and the probability of alpha particles or lithium recoils from the $B^{10}(n,\alpha)Li^7$ reaction reaching the sensitive corona region of the electrode assembly (*2*). Sensitivity is not a function of the total volume of BF_3 gas. The sensitive volume of the spark counter extends only to the alpha-particle range. Our sensitivity is decreased by shielding of additional gas since the anode-wall distance is greater than the alpha range.

The slow-neutron-sensitive spark counter can be converted into a fast-neutron-sensitive spark counter by encasing the detector in a paraffin shield to moderate fast neutrons and covering the exposed surface of the paraffin with a 2-mm cadmium sheet to absorb incident slow neutrons.

Counter Construction

Our counter is enclosed in an airtight stainless-steel cylinder filled with BF_3 to 45 cm Hg (Figs. 1 and 2). The boron in the gas is 96% enriched in B^{10}. The electrode assembly consists of a smooth-surfaced stainless-steel cathode plate 2 in. wide and 3 in. long and an anode array of five parallel 0.005-in. stainless-steel wires. The wires are in a plane 1 mm from the cathode surface and parallel to it.

Between power supply and counter we use a two-resistor series combination to attenuate output pulses from the ~ -200 volts at the anode to ~ -50 volts. A 50-$\mu\mu$f condenser couples the pulse to a scaler.

Slow-Neutron Response

The detector was exposed to a constant flux of slow neutrons from a paraffin-encased 10-mc radium-beryllium neutron source to determine its response as a function of applied voltage. Runs were made at 50-volt intervals between 2,000 and 2,300 volts with and without cadmium shields about the detector. Although the pulse heights were observed to increase with applied voltage, all pulses had identical shapes and magnitudes at any one operating voltage. The results of the measurements are presented in the table. In the absence of neutrons, counting rate was zero over the entire voltage range we studied.

Slow-Neutron Response

Applied voltage	*No cadmium (cpm)*	*Cadmium (cpm)*
2,000	10	0
2,050	30	0
2,100	62	0
2,150	80	0
2,200	115	4
2,250	200	8
2,300	560	75

Gamma Response

With slow neutron source removed, gamma radiation response of the detector was checked by placing a 1 mc Co^{60} needle source in contact with the outer wall of the counter cylinder. The intensity of the gamma radiation at the sensitive region of the electrode assembly was computed to be >500 r/hr. The counter was found to be completely insensitive to this intense gamma radiation.

* * *

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* PRESENT ADDRESS: Physics Department of the University of Maryland, College Park, Maryland.

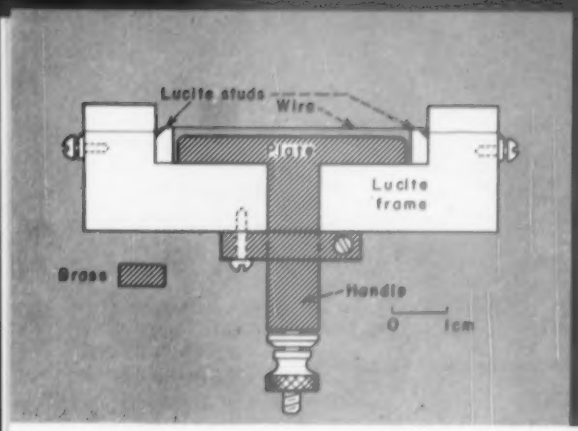

FIG. 1. Single-wire counter has wire stretched in front of metal plate. Sliding handle in frame adjusts separation

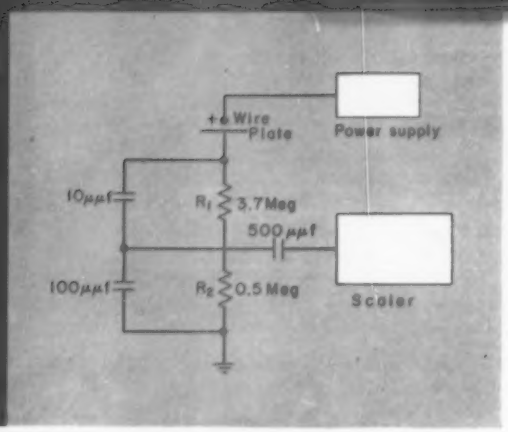

FIG. 2. Circuit used for counting spark pulses has condenser pulse divider. Sum of R_1 and R_2 ($=R$) determines quiescent characteristics

Operating Characteristics of the Spark Counter

This highly specific instrument is sensitive only to alphas. With a boron plate it gives 1 count for 180 neutrons/cm² but < 1 count/hr in a gamma field of 1 roentgen/sec

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ALPHA PARTICLES have low penetrating power and ionize densely. To detect them, therefore, one should have an instrument with a thin window or no window at all, and this instrument should be sensitive to dense ionization and insensitive to the lighter ionization of beta and gamma rays. The Rosenblum spark counter meets both of these criteria. It consists of a wire or a system of wires parallel to a metal plate and maintained at such a potential that the ionization produced by an alpha particle initiates a spark discharge between the plate and the wire or wires. An efficient neutron detector is made by placing a thin layer of boron in front of the wires and detecting the alphas that result from the $B^{10}(n,\alpha)Li^7$ reaction.

Following the original idea of Greinacher (*1*), Chang and Rosenblum (*2*) designed a satisfactory alpha counter that uses the breakdown between a flat metal cathode and a fine wire anode stretched 1.5 mm in front of the cathode. Other workers have improved and modified the spark counter and studied its operation (*3–7*). A standard form has not yet emerged, however, and the mechanism is still obscure in many respects.

Conflicting reports have appeared regarding some of the counter's properties. Connor has observed a decreasing counting rate with increasing applied voltage, while Payne has observed a nearly constant rate. Similar uncertainties exist as to whether the pulse size increases or decreases with applied voltage, whether a positive temperature effect exists, etc.

We have developed single-wire and

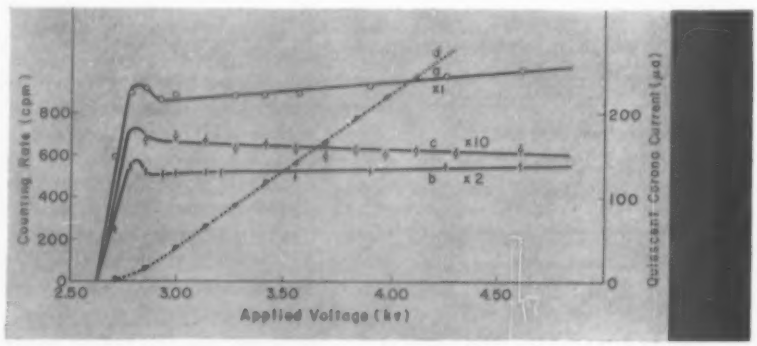

FIG. 3. Counting characteristics (*a*, *b*, *c*) show hump where quiescent current (*d*) starts. With increasing source collimation slope decreases (*a* to *b* to *c*) as oblique rays fail to reach sensitive volume

multiple-wire counters and studied their detection of alpha particles and neutrons. A mechanism has been developed that successfully explains the counting, quiescent-current characteristics, the number-vs-distance curve for an alpha source, and our observed temperature dependence. Some apparently contradictory observations of earlier workers are satisfactorily accounted for.

Single-Wire Counter

The form of our single-wire counter is shown in Fig. 1. A slightly oxidized brass plate and two rectangular Lucite studs rest on the base of a Lucite frame. A 0.1-mm tungsten wire is stretched over the studs and held tight by screws at both ends. The plate and the studs are accurately machined to ensure uniform separation (~1.2 mm) between wire and plate. The plate can be moved by pushing the handle attached to it in and out. The handle can be fixed at any position by the spring-and-screw arrangement.

A positive voltage of 2–5 kv is applied to the wire. The cathode is grounded through a pulse-quenching circuit, and a small part of the positive pulse at the cathode is taken to a Marconi TF 922 scaler for final counting. Figure 2 shows the circuit arrangement used for registering the number of sparking events.

Neutron Detection

Savel (7) was the first to use a modified form of the Rosenblum counter for measuring slow-neutron doses and alpha contaminations. Nearly simultaneously Connor (5) developed an alpha monitor using a multiple-wire system in front of a metal plate. To develop this technique for efficient neutron detection and study of n,α reactions, we undertook construction of a suitable spark counter. Our early experience indicates that we can detect neutrons with some confidence and reproducibility even with as weak a source as 100 mg of radium-beryllium, that is, $\sim 10^7$ n/sec.

We designed a system with 15 wires in front of a 4×4-cm flat brass plate. At first there was some difficulty getting all wires to work together, but by minor adjustments a narrow voltage region was found in which all the wires worked satisfactorily. The counting efficiency was greatly enhanced by using many wires.

Alpha particles released by the n,α reaction from a very thin layer of powdered boron placed close in front of the counter wires were used to detect neutrons. The whole assembly including the neutron source was placed inside a paraffin block of 10-cm wall thickness. With the 100-mg Ra-Be source on a line perpendicular to the plate, 7 ± 1 cpm was observed. When the boron layer was removed, no background counting was observed over a period of about an hour, even in the presence of the intense gamma rays (~1 r/hr) from the source. In our geometry one observed pulse roughly corresponds to the passage of 20 n/cm²/sec in the counter area. It seems, however, that a further improvement in the efficiency of neutron counting is possible by improving the counter design and using a boron-coated cathode. Work is in progress in this direction.

Counting Mechanism

Our mechanism for the counting action can be described in terms of the spark-counter arrangement of Fig. 2. A strong electric field exists between the wire anode and the plate cathode separated by a distance of the order of 1 mm in atmospheric air. In the absence of any strongly ionizing particle passing between the electrodes and for a sufficiently strong electric field, a quiescent corona current starts and with increased field is ~100 μa. This indicates the presence of a fairly dense space-charge sheath forming a small distance from the wire in the direction of the plate. Most of the properties of the spark counter seem to depend largely on the behavior of this space-charge sheath under various circumstances. This is a situation that does not exist in the G-M counter, where there is practically no quiescent corona current.

There is an external resistor $R = R_1 + R_2$ in series with the electrodes, and a corresponding drop RI when a corona current I flows. The effective potential drop between the electrodes is therefore

$$V_{eff} = V_a - RI$$

where V_a is the applied potential. It is therefore clear that the effective electric field between the electrodes, which determines the counting behavior, is partly dependent on the external resistor. As the applied voltage is varied the corona current I also changes, and therefore V_{eff} may not change in the same way as V_a.

Suppose an ionizing particle is now allowed to pass between wire and plate. On account of the strong electric field in the neighborhood of the wire an electron avalanche starts at once in the same region. But this will succeed in producing a spark breakdown between

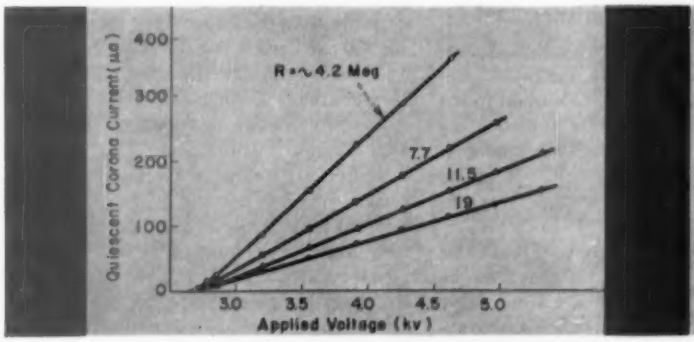

FIG. 4. Plots of I vs V_a for different values of external resistance R show that corona sheath obeys equation, $V_{eff} = V_a\,[1 - k(R)R]$

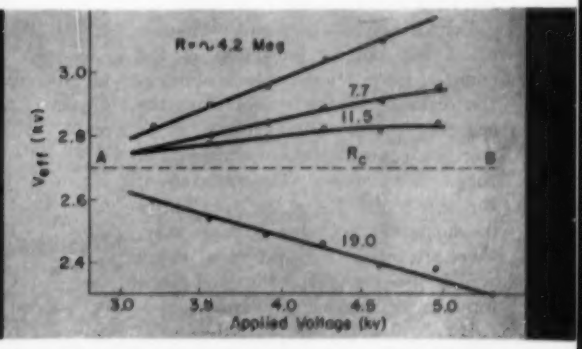

FIG. 5. When V_{eff} is computed from Fig. 4, result indicates that for a critical value of R, V_{eff} should be constant with changing V_a.

the electrodes only if the avalanche is sufficient to overcome the strong negative space charge present near the wire. This explains the immunity of the spark counter to beta and gamma radiations, which produce a much weaker avalanche than does a heavier ionizing particle like the alpha particle.

On further increase of V_a, the *a priori* probability for an avalanche to produce a spark breakdown, or *a priori* sensitivity, increases, causing an increased number of alpha particles from the source to be counted. It may happen that when V_a has increased beyond a certain limit, the corona-space-charge effect, which also increases with V_a, is also very high. This might lead to a reduction in *a priori* sensitivity, since the limited specific ionization of some

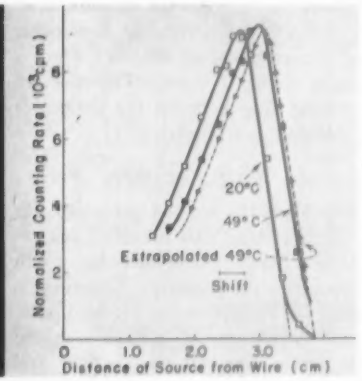

FIG. 6. Solid curves are counts observed at 20° C and 49° C as Po^{210} source approaches counter. Dotted curve is shifted version of 20° curve

of the particles may not be sufficient to overcome the space charge and may thereby fail to produce a breakdown. We can then expect a slight fall in the counting rate although V_a is rising.

For a heavily ionizing particle like an alpha the specific ionization in the counting space depends somewhat upon the distance of the source from the counter. Sparking breakdown requires a critical concentration of ionization in the sensitive space. The variation of the source distance or the direction of incidence of the alpha particle may, therefore, cause a variation in the *a priori* sensitivity, giving the spark counter strong directional properties.

Let us now examine the effect of R and V_a on V_{eff} and the counting rate of the counter. It can be seen from theoretical consideration that for a given value of R, an approximately linear quiescent-current characteristic exists within a certain range of V_a; that is, as V_a increases, I also increases linearly. The magnitude of I is, however, determined by both the value of R and the dynamic resistance of the space-charge sheath. On increasing R for a given V_a, the current I may therefore decrease nonlinearly with R. We may thus represent I as

$$I = k(R)V_a$$

where $k(R)$ is the constant slope of the I vs V_a characteristic. The quantity $k(R)$ is a constant for a given R. Substituting in our earlier relation, one gets

$$V_{eff} = V_a[1 - k(R)R]$$

From this equation it follows that for values of $k(R) < 1/R$, V_{eff} will increase with V_a. This will be true only for certain values of R. For some other (larger) values of R, $k(R)$ may exceed $1/R$, giving a reduction in V_{eff} with increasing V_a. If the electrode spacing is small, $k(R)$ can exceed $1/R$ even for moderate values of R. The consequence of this effect will be a decreasing *a priori* sensitivity and a decreasing counting rate with increasing V_a. For a given electrode spacing there is therefore a critical resistor value R_c above which the counting rate will decrease with increasing V_a. Further, R_c will decrease with decreasing electrode spacing.

Experimental results on counting and quiescent-current characteristics, the number-distance curve for polonium alphas, the temperature coefficient of counting, and the detection of neutrons can be explained by the counter mechanism outlined. Some of the apparently conflicting results of earlier workers can also be accounted for.

Counting Characteristics

A polonium source on a 1-cm-diameter thin platinum disk was placed facing the counter on a line normal to the plate. The variation of the counting rate with applied voltage was observed. Figure 3 shows the result. It differs from what earlier workers have observed (*3–6*).

The difference between our result and those of Connor and Payne can be understood as follows. Alpha rays from our extended polonium source enter the counter at many different angles. Since with increase in V_a, *a priori* sensitivity gradually increases, some of the oblique rays that could not produce sparking breakdown at lower voltages are counted at higher voltages. This gives rise to the small positive slope of our counting characteristic. If the source is highly collimated, the directional effect on the counting rate will be practically absent, but the effect of increasing space charge and that of the external resistor as explained previously will still remain. These may result in a positive, zero, or negative slope of the counting characteristic, depending on the particular circumstances of the experiment.

It appears to us that the conditions and geometry of Connor's counting arrangement (*4*) were similar to ours except that he used a smaller separation between the electrodes. The latter might be responsible for a strong space-charge effect at higher voltages. Also, the external resistor used in his case might exceed the critical value R_c for his small electrode spacing. These two effects might combine to give a slow decrease in the counting rate with increasing voltage. Exact reasons can not be assigned without knowing the quiescent-current characteristics obtained in Connor's experiment. The nearly constant rate observed by Payne (*6*) can also be similarly understood in a general way.

To investigate the effect of source collimation we observed counting characteristics with two different collimating tubes of 1.4- and 0.6-mm diameters mounted over the polonium source. Figure 4 shows the results. It is clear from the three plots that the slope of the counting characteristic progressively decreases on putting the collimators in front of an extended source and even becomes negative for a highly collimated source. Clearly the absence of oblique rays for a collimated source reduces the slope. The increasing V_{eff} can even make the slope negative, as is probably the case for the lowest curve. The middle curve represents perhaps an intermediate case with limited collimation. The observed negative slope of the high-collimation curve is not the effect of the external resistor in our case, as will be clear from discussion further on, but should be attributed to the collimation and space-charge effects.

When examined on an oscilloscope, the counter pulses were found to be of more or less the same magnitude for a fixed applied voltage and the pulse-

rise time $<10^{-7}$ sec, but the pulse height increased by about 10% when the voltage was raised from 2.6 to 4.6. Of this increase about half took place between 2.6 and 2.9 kv before the start of the plateau region. The rest occurred over the entire plateau region; the quiescent corona current increases with voltage, so that the effective field between the wire and the plate increases only very slightly. Indeed, depending upon the external resistor used, V_{eff} can even decrease with increasing V_a in certain cases, where a fall in the pulse heights can be anticipated.

Quiescent Current

A corona discharge is set up around the anode wire as the applied voltage is raised above the counting threshold. A bluish glow in the form of a fine streak of light is observed all along the wire's length. The visible glow extends about one wire diameter toward the plate.

With the source removed we studied the variation of corona current with applied voltage by placing a microammeter in series with the cathode resistor ($R = R_1 + R_2$, Fig. 2). The quiescent-current characteristic for the counting arrangement described earlier is shown in Fig. 3. It is seen that the quiescent current sets in at the beginning of the hump in the counting characteristic. Then there is a very small region where it increases nonlinearly with increasing applied voltage. Beyond about 2.9 kv it increases linearly. The start of the linear region corresponds clearly to the start of the counting plateau.

Examining the quiescent condition on an oscilloscope, we observe corona pulses, their number increasing with the applied voltage. Near the applied voltage corresponding to the hump in Fig. 3, corona pulses of comparatively large magnitude are observed and corona oscillations were loud enough to be clearly audible. Probably the slight increase in the counting rate at the hump is due to these corona oscillations. Once the corona discharge has extended over the whole length of the wire, the counting condition seems to stabilize and the observed rate settles down to the normal value.

The quiescent current characteristics were also observed for four different values of R (4.2, 7.7, 11.5, and 19 megohms). The results are shown in Fig. 4. These curves, obtained for different values of R, fit well with the ideas outlined earlier. For each of these curves, V_{eff} can be calculated easily and the previous predictions regarding the variation of V_{eff} with V_a examined. The plots of V_{eff} vs V_a obtained in this way are given in Fig. 5. Clearly the higher curves correspond to the case when $k(R) < 1/R$ and V_{eff} increases with V_a. As expected, the rate of increase of V_{eff} with V_a gradually falls as R is increased. The bottom curve corresponds to the case where $k(R) > 1/R$ ($R \cong 19$ megohms), and here V_{eff} decreases with increasing V_a. For R between 11.5 megohms and 19 megohms, there should also be a critical resistance R_c for which V_{eff} would remain practically constant (dotted line AB) over a large range of V_a. For all values of R greater than R_c, V_{eff} and hence the counting rate are expected to decrease with the applied voltage, as has been actually observed by Connor.

For all our subsequent measurements we have a total quenching resistance $R = 4.2$ megohms.

Rate vs Distance

The number-distance curve for an alpha source was measured with this counter using a collimated old polonium source. Figure 6 shows the number-distance relation at room temperature. The shape of the curve is similar to that observed with other alpha detectors near the end of the range. The considerable spread of the curve observed between the peak and the maximum distance at which counts were registered ($\sim$3.6 cm) is due to our using a fairly old polonium source (*8*). With the source nearer to the counter there is observed, however, a remarkable fall in the counting rate with decreasing distance, making the monochromatic alpha peak singularly fine. This sudden fall seems to arise from two causes. First, the specific ionization of the alpha particles is known to fall somewhat for distances less than those corresponding to the maximum of the Bragg curve. This will amount to a reduction in *a priori* sensitivity and a consequent fall in the counting rate as the source approaches the counter. However, this factor does not seem to account for the rather steep fall observed. The increase in the number of oblique rays as the source is moved toward the counter seems to be the dominant cause, the *a priori* sensitivity being lower for the oblique rays than for the normal rays for the same working voltage. The range, however, of monochromatic alpha particles from a freshly prepared source can be determined by this counter with reasonable accuracy.

Temperature Dependence

None of the earlier workers have tried to study the temperature dependence of the spark counter. Connor has expressed the view that it is likely to be slight, but no exact data on temperature effect appear to exist.

To study the temperature dependence it is essential to allow for the apparent change in the range of particles caused by the change in the density of the surrounding air as its temperature is changed. We thought it best to study the temperature dependence by observing the number-distance curve for the polonium source at two different temperatures. The first observation was at room temperature (20° C). Then the counter was placed in a thermostatically controlled volume where the temperature was maintained at 49 $\pm$ 1° C. The results are shown in Fig. 6, where one solid curve corresponds to room temperature and another to the higher temperature. The dotted curve was drawn by shifting the room-temperature curve toward higher d values by an amount equal to the apparent increase in the range of the alpha particles expected at the higher temperature. The dotted curve may be called the extrapolated number-distance curve for the higher temperature.

It will be seen from the figure that a better fit between the shifted curve and the measured one might result from a simple "stretching" of the Bragg curve; that is, by increasing all abscissas by the same fraction of themselves so that extrapolated ranges (or the maxima) would coincide. We have not attempted this slightly more complicated analysis, but it appears that a small temperature effect exists.

* * *

We are greatly indebted to Professor D. S. Kothari for stimulating this work and his kind and continued interest during its progress. This work was carried out under the financial support of the Department of Atomic Energy, Government of India.

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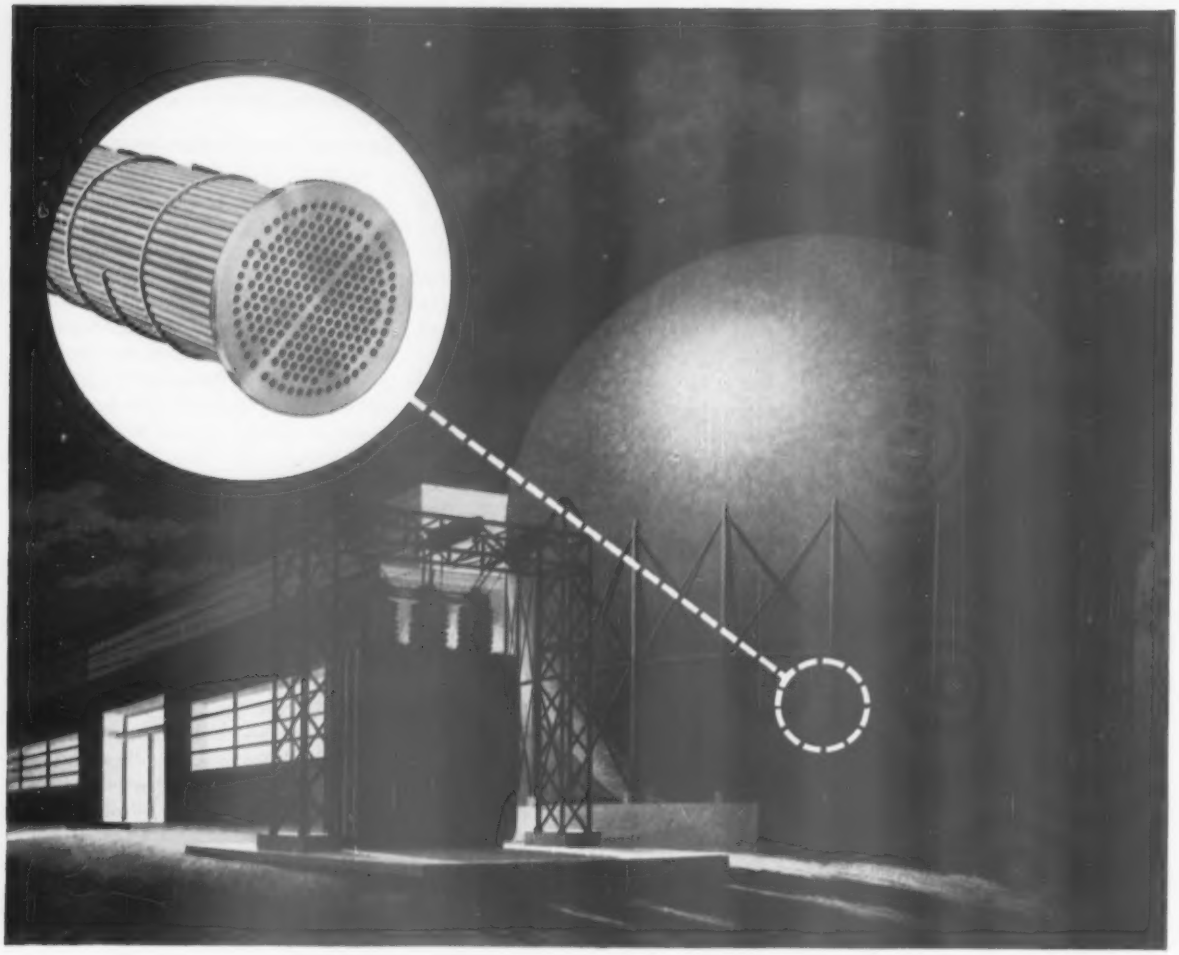

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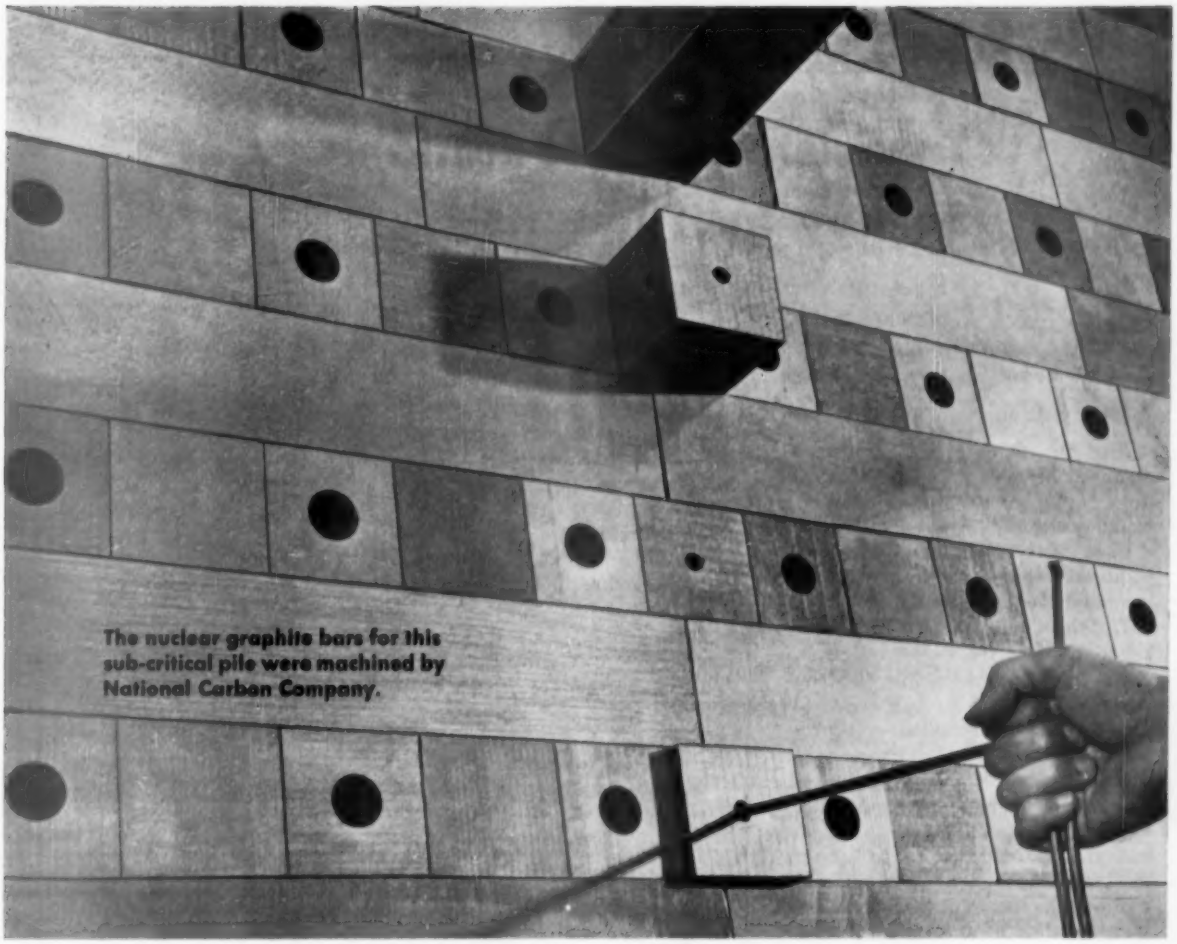

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Transistorized Radiation Survey Instruments

By W. G. SPEAR
Hanford Laboratories
Hanford Atomic Products Operation
Richland, Washington

The radiation-survey instruments described here are designed for lightness, compactness and ease of maintenance.

Each instrument was tested at temperatures from −20° F to +160° F to assure satisfactory operation.

The alpha scintillation meter weighs 2.5 lb, as does the G-M-tube meter. The combination alpha and G-M instrument weighs 3.5 lb, the gamma and the beta-gamma scintillation instruments weigh 4 lb each, and the neutron-survey instrument weighs 5 lb. All of the instruments represent an extended battery life and an approximate weight and volume reduction of 50% from most of their commercial counterparts. The neutron survey meter weighs less than one-fourth as much as typical available neutron-detection instruments with moderator.

Extensive use of commercially available transistors and associated printed circuit boards resulted in a reasonably complete assortment of ratemeter type of portable survey instruments. With the high-voltage supply and the counting-rate-meter circuit common to all instruments and with the transistor amplifier, where necessary, common to all but one instrument, interchange of complete circuitry is possible with resultant cost and inventory reductions. Eventually, all survey instruments could be standardized including the photomultiplier tubes with only a change in detectors necessary for versatile surveying of alpha, beta, gamma contamination and neutron flux.

Detectors

The alpha detector is a ZnS-coated Lucite plate and an 1P21 photomultiplier tube; plutonium contamination of 100 dpm over an 8-in² area is detectable with <1 cpm background. For beta-gamma detection, an anthracene crystal with a 6199, 6291 or 6292 crystal or a thin-wall G-M tube is used; for strictly gamma detection, a NaI(Tl) crystal provides a gamma sensitivity of 0.01 mr/hr. A LiI(Eu) crystal is used for thermal neutrons and surrounded by ceresine wax moderator for intermediate neutrons. Lucite with dispersed ZnS powder is used for fast neutrons. Sensitivities are 450 cpm per n/cm²/sec for the LiI(Eu) crystal and well moderated Ra-Be neutrons and 20–25 cpm/fast neutron/cm²/sec.

High-Voltage Supply

The same type of high-voltage supply (Fig. 1) is used in all the instruments.

The supply consists of a single General Electric type 2N43 junction transistor in a blocking oscillator circuit. The circuit may be used to produce either positive or negative 900–1,200 volts by properly connecting the transformer secondary winding, the rectifier

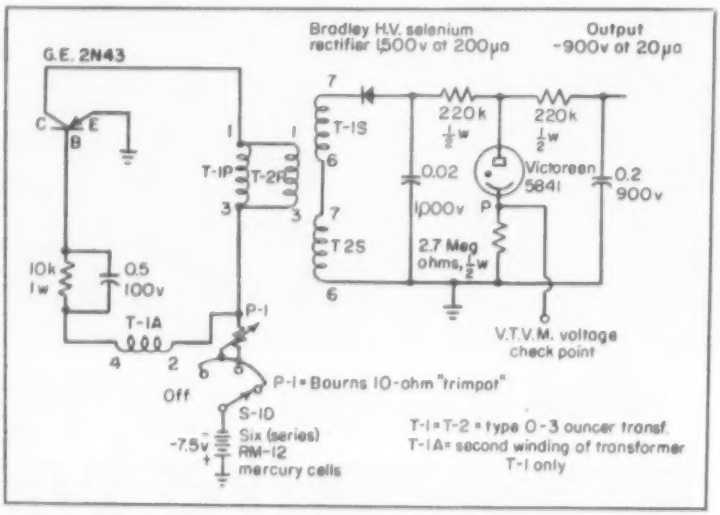

FIG. 1. Transistorized power supply is used for all the survey instruments

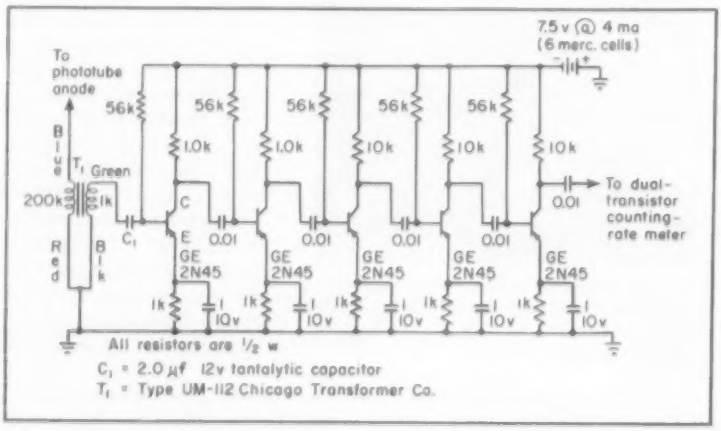

FIG. 2. Five-transistor amplifier is adequate for α, β, and γ detection

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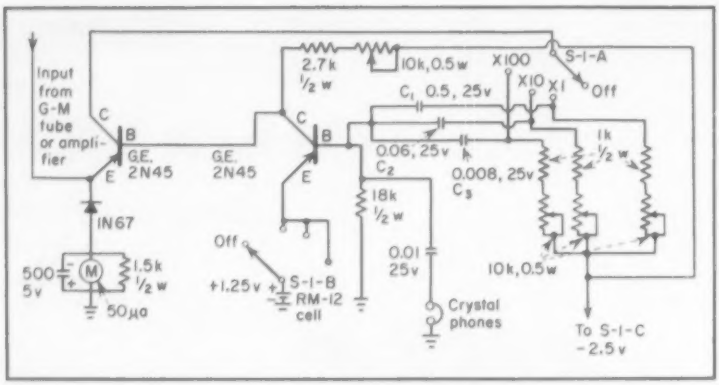

FIG. 3. Transistorized ratemeter circuit is used in all the units

and the voltage regulator. For 900 volts, a Victoreen 5841 regulator is used, and Victoreen VXR-1000 and 6143 regulators are used for 1,000 and 1,200 volts, respectively.

A repetition rate of about 350 pulses/sec with a pulse width of 300 μsec was optimum, and the maximum conversion efficiency with a 20-μa high-voltage load current is about 38% with a resultant battery life of approximately 300 hr of continuous operation. The 10-ohm rheostat is used to vary the input current for most efficient operation, depending on the transistor used.

The complete high-voltage supply, including the miniature rheostat, is enclosed in a steel can to prevent regenerative pickup by the amplifier. Negative high voltage is used with the photomultiplier tubes to eliminate the need for high-voltage capacitors from the photomultiplier anode to the amplifier. For 1,000- and 1,200-volt operation, rated 1,600-volt capacitors are used in the high-voltage filter circuit.

Amplifier

The transistor amplifier shown in Fig. 2 utilizes five General Electric type 2N45 transistors and is used in all but two of the instruments.

For neutron detection, a subminiature three-stage vacuum-tube amplifier is used to obtain the necessarily precise discrimination. For alpha, beta and gamma contamination detection, the five-transistor amplifier is adequate.

A 5-mv input pulse to the amplifier produces 5-volt constant-amplitude output pulses to drive the counting-rate-meter circuit. The total d-c input power for the amplifier is only 30 mw from the six series 1.25-volt mercury cells, giving a battery life of >400 hr.

The amplifier has been fabricated in printed-circuit form on a 2.5×3-in. plug-in board. Such a system permits rapid interchange of complete circuits, and this increases the available field time for the instruments.

A three-stage subminiature vacuum-tube amplifier used in the neutron survey meter will be supplanted by a linear transistor amplifier when the higher-frequency transistors become readily available.

Ratemeter Circuit

The counting-rate-meter circuit is shown in Fig. 3. This circuit, used in all of the instruments, uses two General Electric 2N45 junction transistors connected in a bistable circuit.

In the operation of the circuit, the second transistor is conducting when no pulses are being received by the input transistor. This quiescent conduction is sufficient to bias the base of the input transistor to a point of nonconduction. An incoming pulse to the emitter of the first transistor, which presents a high input impedance when not conducting, starts conduction and the collector of the first transistor becomes less negative. This change is applied to the base of the second transistor causing it to cease conduction. The range capacitor (C_1, C_2 or C_3) charges through the base resistor of the second transistor, and when the base voltage nears zero, the regenerative action reverses and the second transistor again conducts, thus causing the first transistor to again cease conduction. The circuit is then ready to accept the next pulse.

For use with metal-walled G-M tubes using positive high voltage and consequently producing negative pulses, the counting-rate-meter circuit uses N-P-N General Electric 2N168 transistors. The circuit used is the same as described except for battery and diode

reversal and signal application to base of the second transistor.

The counting-rate ranges in both cases are 1,000, 10,000 and 100,000 cpm, full scale. The time constants for the respective ranges are approximately 4, 3 and 2 sec. The d-c drain from the 2.5-volt battery is about 1.5 ma, giving a battery life of >800 hr.

The counting-rate-meter circuit was fabricated on a plug-in printed circuit board to facilitate maintenance.

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Resolving Complex Decay Curves

By D. H. PERKEL*
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Tracerlab, Inc.
Richmond, California

The customary method of analyzing decay curves of samples containing several components is to allow shorter-lived components to decay out until only the longest-lived one remains. The straight line on a plot of logarithm of observed activity vs time, representing the longest-lived component, is extrapolated backwards and subtracted from the total curve. The resulting difference curve is plotted, and the next most long-lived component is subtracted in a similar manner. Usually, no more than three components can be resolved by this method (*1*).

The subtraction method is most useful when (a) there is a wide range in half-life values, (b) there are a great many decay points, for which graphical methods represent a distinct saving in time over numerical methods, (c) the activity level is high enough to permit a long decay curve to be taken, and (d) half-lives are not known or not known accurately.

The subtraction method cannot be applied, however, when only a few decay points are known, or if the curve has not been taken to the point where the shorter-lived components have died out. The "linear" method described in this article is an analytical method of resolving complex decay curves, which can be applied to a limited decay curve. The method involves a relatively large amount of computation, and the half-lives of at least the shorter-lived components must be known accurately.

This linear method is fairly well known (*2–4*), but has been applied mainly to two-component decays. In this report it is shown how the method can be extended to three-component decays, and can detect and measure contamination in a two-component sample.

Two-Component Curve

If the observed activity† $R(t)$ at time t (from zero time) is the sum of two independently decaying components A and B, then its value is given by

$$R = A_0 e^{-\lambda_a t} + B_0 e^{-\lambda_b t} \qquad (1)$$

where the zero subscript refers to zero-time activities. Let $\lambda_a > \lambda_b$.

When multiplied through by $e^{\lambda_b t}$, Eq. 1 becomes

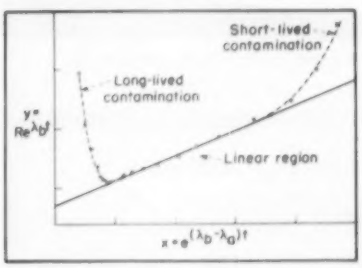
$$Re^{\lambda_b t} = A_0 e^{(\lambda_b - \lambda_a)t} + B_0 \qquad (2)$$

This equation is of the form

$$y = mx + b \qquad (3)$$

where x and y are known, and the slope m and the y-intercept b are equal to the zero-time activities A_0 and B_0, respectively.

In practice, $Re^{\lambda_b t}$ is plotted vs



TYPICAL TWO-COMPONENT linear plot, showing effect of contamination

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† It is assumed throughout that all observed activities R are known to the same proportional accuracy.

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$e^{(\lambda_b-\lambda_a)t}$ (on linear scales) for each point on the decay curve. The slope and y-intercept of the resulting straight line are obtained visually or by a least-squares computation.

Solution of the well-known least-squares equations gives:

$$A_0 = m = \frac{n\Sigma xy - \Sigma x\Sigma y}{n\Sigma x^2 - (\Sigma x)^2}$$

$$B_0 = b = \frac{\Sigma y\Sigma x^2 - \Sigma x\Sigma xy}{n\Sigma x^2 - (\Sigma x)^2} \quad (4)$$

where $x = e^{(\lambda_b-\lambda_a)t}$, $y = Re^{\lambda_b t}$, n = number of observations.

Two Components Plus Contamination

Here, it is assumed that

$$R(t) = A_0e^{-\lambda_a t} + B_0e^{-\lambda_b t} + C(t) \quad (5)$$

where $C(t)$ is the time-dependent contaminating activity. Letting $x = e^{(\lambda_b-\lambda_a)t}$ and $y = R(t)e^{\lambda_b t}$ as before, Eq. 5 can be rearranged to give

$$y = mx + b + C(t)e^{\lambda_b t} \quad (6)$$

Let y be plotted vs x as before. If $C(t)$ has an effective half-life significantly shorter than that of either known components, then, at large values of t, $C(t) \ll R(t)$, and $y \approx mx + b$. This straight line is well defined in the plot. The departures of y from the extrapolation of this line to small values of t are given by the last term in Eq. 6, $C(t)e^{\lambda_b t}$. Similarly, if the contaminating activity has a half-life significantly longer than that of either of the two known components, then the plot of y vs x will be linear at small values of t and will depart from linearity by amounts $C(t)e^{-\lambda_b t}$ for large values of t. These relationships are shown in the illustration.

In either of these two cases, then, there exists a linear portion of the curve. This linear portion may be extrapolated to the nonlinear region, and the departures are a function of the contaminating activity, viz., $C(t)e^{\lambda_b t}$. If m and b have been found (as by least-squares computation for the linear portion alone), then the contaminating activity is given simply by

$$C(t) = [y - (mx + b)]e^{-\lambda_b t} \quad (7)$$

The zero-time activities A_0 and B_0 are, as before, given by m and b respectively.

Since $C(t)$ is time-dependent, it will display a characteristic decay. If it consists of one component, the half-life can frequently be found readily, and the contaminant identified thereby.

If, however, the contaminating activity has an effective half-life not

exactly known but intermediate between the half-lives of A and B, then the method of this section cannot be used.

Three-Component Curve

Here, it is assumed that

$$R(t) = A_0 e^{-\lambda_a t} + B_0 e^{-\lambda_b t} + C_0 e^{-\lambda_c t} \quad (8)$$

(without loss of generality, we may let $\lambda_a > \lambda_b > \lambda_c$), and hence

$$Re^{\lambda_c t} = A_0 e^{(\lambda_c - \lambda_a)t} + B_0 e^{(\lambda_c - \lambda_b)t} + C_0 \quad (9)$$

This is of the form

$$z = ax + by + c \quad (10)$$

the equation of a plane, where

$$\begin{aligned} z &\equiv Re^{\lambda_c t} \\ x &\equiv e^{(\lambda_c - \lambda_a)t} \\ y &\equiv e^{(\lambda_c - \lambda_b)t} \end{aligned} \quad (11)$$

Although any three points on the decay curve (R vs t) can yield values for a, b and c ($a = A_0$, $b = B_0$, $c = C_0$) by simultaneous solution of the equations

$$\begin{aligned} z_1 &= ax_1 + by_1 + c \\ z_2 &= ax_2 + by_2 + c \\ z_3 &= ax_3 + by_3 + c \end{aligned} \quad (12)$$

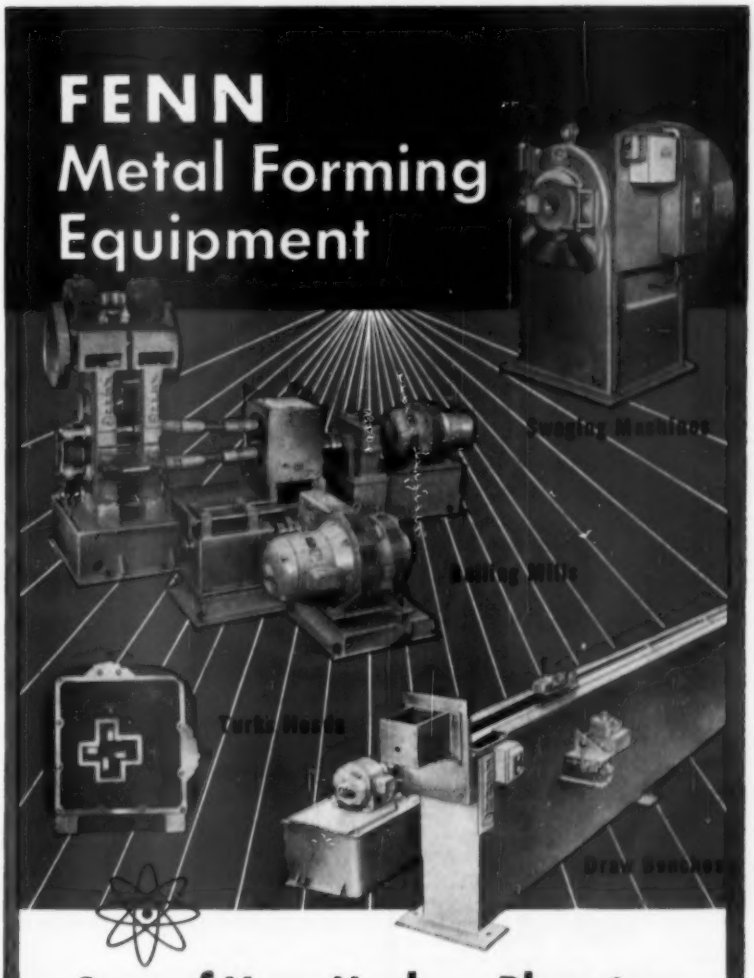
in practice any three consecutive points can be expected to lead to subtraction of numbers of like magnitude, with consequent loss of precision. There are two methods of solution: (a) several widely spaced triads of points may be chosen, each solved simultaneously for the three components, and the several values for each component averaged, or (b) a least-squares method may be used.

The three numbers to be determined represent the intercepts of the plane described by Eq. 10. Each point on the decay curve yields one point near the plane, and the "best" plane, according to the principle of least squares, has the intercepts A, B and C, which are given implicitly by the equations*

$$\begin{aligned} A\Sigma x + B\Sigma y + Cn &= \Sigma z \\ A\Sigma x^2 + B\Sigma xy + C\Sigma x &= \Sigma xz \\ A\Sigma xy + B\Sigma y^2 + C\Sigma y &= \Sigma yz \end{aligned} \quad (13)$$

These equations can be solved explicitly by well known methods, such as successive eliminations, ratios of determi-

* Equations 13 were obtained by minimizing the sum $\sum_n (Z_i - s_i)^2$, where Z_i is defined as $Ax_i + By_i + C$, the z-component of the "best" plane. Only the z-component deviations were used since z (and not x or y) is a function of the observed counting rate.



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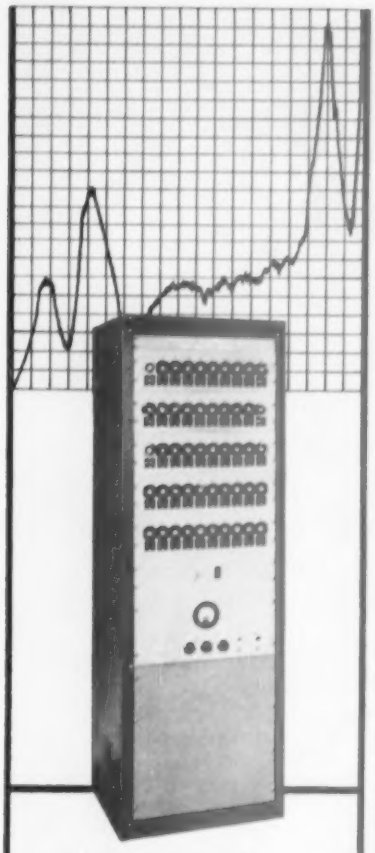

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nants, or the Crout-Milne matrix method (*5*).

It must be emphasized that this method for three-component resolution is limited to cases where the decay constants of the activities are known accurately, and where the contribution of a fourth (contaminating) activity can be assumed, with reasonable assurance, to be negligible.

A refinement of this method to incorporate weighting factors based on the statistical accuracy of the various points on the decay curve is straightforward, but the computational labor in applying this refinement would increase tremendously. This objection can, of course, be overcome if the services of an electronic digital computer are available.

* * *

The author wishes to acknowledge the generous assistance of Donald L. Forrest, who reviewed the manuscript and offered several valuable suggestions.

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Tests of Decontaminants for Steel, Lead and Glass

By JOHN B. HUFF

Atomic Energy Division, Phillips Petroleum Company, Idaho Falls, Idaho

A decontamination problem arose with the lead and stainless steel sample pigs used for transporting highly radioactive samples at the Idaho chemical processing plant. Spilled samples had contaminated the pigs to the extent that they showed contact readings exceeding 1 r/hr.

A reasonably satisfactory electrodecontamination process was developed (*1*). The results of the studies leading up to this method are presented here and should be of use for similar problems.

Sample-Disk Preparation

Dissolver product (nitric acid solution of fuel elements) from the chemical processing plant was chosen as the test contaminant. It was diluted with 1*M* nitric acid and the same sample was used for all tests. The fission products in the liquid sample were estimated at <1 ppm. The radioisotopes deposited on the sample disks were thus at tracer level and probably undetectable by usual analytical methods.

Stainless-steel counting disks, 48 mm in diameter, polished on one side, were prepared. The disks were usually precleaned with acetone. A 30-mm retaining circle was inscribed on the polished side with wax pencil and templet. One-milliliter samples were evaporated within the circle under a heat lamp.

Lead disks were cut from lead sheets. Those with obvious surface blemishes were discarded. Glass samples were prepared on 26-mm-diameter watch glasses. The concave side received 1 ml of solution.

Each sample was rinsed with tapwater after treatment. They were then dried by evaporation, except where wiping is specified in the table on the facing page.

The samples were counted with an end-window G-M tube in a lead shield at $\sim$10% geometry. A jig provided counting reproducibility. The values given in the table are average net cpm from 5-min counts. A background count was made each time samples were counted. The counter was probably insensitive to some low-energy betas.

Part of the variation in initial counts given in the table is due to decay, since the liquid sample was used for about a year. Prepared disks stored in envelopes were also subject to mechanical loss of contaminant.

Discussion

Electrolytic methods are at the top of the lists for both stainless steel and lead. However, these methods are subject to some operational difficulties. First, there is the problem of getting the desired current density on internal surfaces, such as the working parts

Procedures Tested for Decontaminating Stainless Steel, Lead and Glass

	*Method**	*Initial count*	*Final count*	*Decontamination factor (initial/final)*
Stainless steel				
1	Anode 4 min in hot dil H_2SO_4, 150 amp/ft²	15,562	Bkgd	16,000
2	Anode 2 min in hot Oakite 90, 150 amp/ft², wire brushed with dil H_2SO_4	14,652	1	15,000
3	Boiled 5 min in 1*M* NaF, then anode 3 min, 100 amp/ft², same soln	9,970	Bkgd	10,000
4	Soaked 30 min in warm 20% HNO_3 by vol plus 3% NaF by weight	9,773	Bkgd	9,800
5	Boiled 20 min in soln Turco, 45 gm/l	9,690	Bkgd	9,700
6	Boiled 30 min in soln Turco, 45 gm/l	9,776	5	2,000
7	Boiled 70 min in soln Turco, 45 gm/l	11,623	6	1,900
8	Boiled 40 min in soln Turco, 45 gm/l	9,850	6	1,600
9	Anode 2 min in soln Turco, 45 gm/l	5,509	7	790
10	Boiled 20 min in 0.25*M* H_3PO_4	4,246	7	600
11	Boiled 20 min in soln Turco, 45 gm/l (sample prepared on lacquered disk)	2,620	12	220
12	Boiled 40 min in soln Green Detergent, 45 gm/l	9,846	58	170
13	Boiled 30 min in 2*M* citric acid	10,018	84	120
14	Boiled 20 min in soln Green Detergent, 45 gm/l	9,288	88	110
15	Boiled 30 min in 1*M* HNO_3-0.003*M* periodic acid	9,855	91	110
16	Boiled 30 min in 1*M* HNO_3	10,174	121	84
Lead				
1	Anode for 1 min in hot 1*M* KOH followed by hot rinse in 1*M* HNO_3; several cycles; 150 amp/ft²	15,460	Bkgd	16,000
2	Anode in alkaline soln potassium tartrate for 30 sec followed by 60-sec soak in same soln; 150 amp/ft²; 4 cycles	15,450	Bkgd	15,000
3	Boiled 1 hr in soln Turco, 45 gm/l, counted 1,309 cpm; anode in hot 1*M* HNO_3 for 1 min	17,310	12	1,400
4	Anode in Oakite 90, 60 gm/l, 5 min at 75 amp/ft²; wiped with tissue; counted 1,034 cpm; soaked 2 min in cold 1*M* HNO_3; wiped with tissue	17,363	34	510
5	Boiled 90 min in soln Versene, 45 gm/l; counted 147 cpm; soaked 2 min in hot dil H_2SO_4, dried with rags	17,365	83	210
6	Soaked 20 min in hot 1*M* HNO_3	11,371	79	140
7	Anode 2 min in soln Turco, 45 gm/l, 70° C, 20 amp/ft²; boiled 20 min in 2*M* citric acid	10,231	108	100
Glass				
1	Boiled 20 min in 1*M* tartaric acid	9,878	Bkgd	9,900
2	Boiled 20 min in soln Green Detergent, 45 gm/l	9,953	Bkgd	9,900
3	Boiled 20 min in soln Turco, 45 gm/l	9,897	Bkgd	9,900
4	Boiled 20 min in 1*M* citric acid	10,315	9	1,100
5	Boiled 20 min in 1*M* HNO_3	10,260	12	850
6	Boiled 20 min in soln Versene, 45 gm/l	10,082	12	830
7	Boiled 20 min in soln Na_2CO_3	9,780	483	20
8	Boiled 20 min in 1*M* NaOH	10,020	2,946	3

* The following cleaning compounds are used:

Calgonite—glassy sodium phosphate sold by Calgon, Inc., Pittsburgh, Pa.

Green Detergent—liquid cleaning solution sold by Decker Brothers, 256 Broadway, Idaho Falls, Idaho.

Oakite 90—alkaline electrocleaning compound sold by Oakite Products, Inc., 19 Rector St., New York 6, N. Y.

Turco—Turco 4182-A, laundering compound sold by Turco Products, Inc., Los Angeles 54, Calif.

Versene—disodium salt of ethylene diamine tetraacetic acid, sold by Versenes, Inc., Framingham, Mass.

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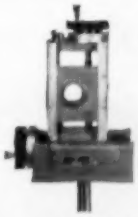

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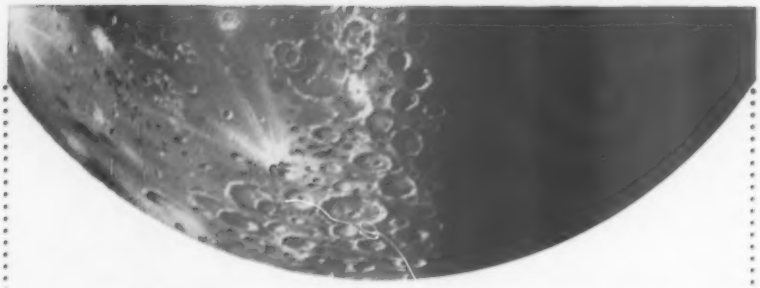

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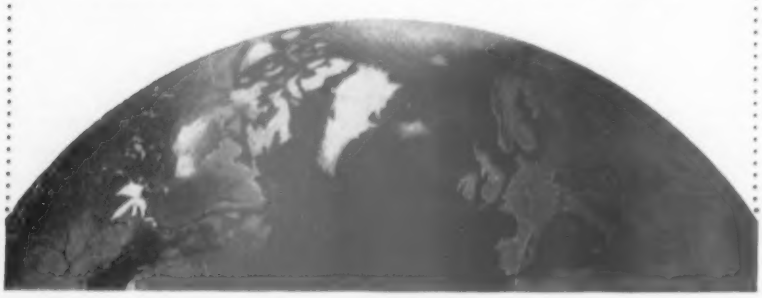


within a sample pig. Secondly, reagents in these methods noticeably etch stainless steel. The corrosion of the article being decontaminated is usually negligible. However, if the spent solutions are sent to permanent storage, the presence of fluoride or sulfate ions is considered a corrosion hazard.

It was observed that lead anodes in alkaline solutions lost metal at a rapid rate, but the stain left by the drying sample tended to remain. This residue, which was assumed to be a complex mixture of lead oxides, was refractory. It may be reasoned that the anodic treatment changed the lower oxides to higher, more soluble ones. The sixth procedure caused a rapid dissolution of the lead, but left considerable contamination.

Suggested procedures. From these tables several procedures can be devised for decontaminating stainless-steel-lead assemblies. In general, anodic treatment in a hot alkaline solution containing some alkali tartrate is recommended for the first step. This step is not especially effective for removing fixed contamination, but it is an easy way to remove adhering contaminated soils. An acid rinse containing tartaric, citric, acetic or nitric acids is preferred over sulfuric acid. The third step might be directed toward decontaminating the stainless steel by boiling in Turco* or phosphoric acid solution. Recycling is frequently necessary.

Lead is so soft that some of the surface can be removed by wiping with a cotton cloth; exposed surfaces can be finished by this method. Rubbing or polishing may be profitably employed as a final step on the unobstructed surfaces of most metals.

Decontamination presents many unique problems, so that a standard approach is not recommended. If the work is contaminated to the extent that a curie of activity, for example, must be removed, the disposal of wastes and confining of contamination will proba-

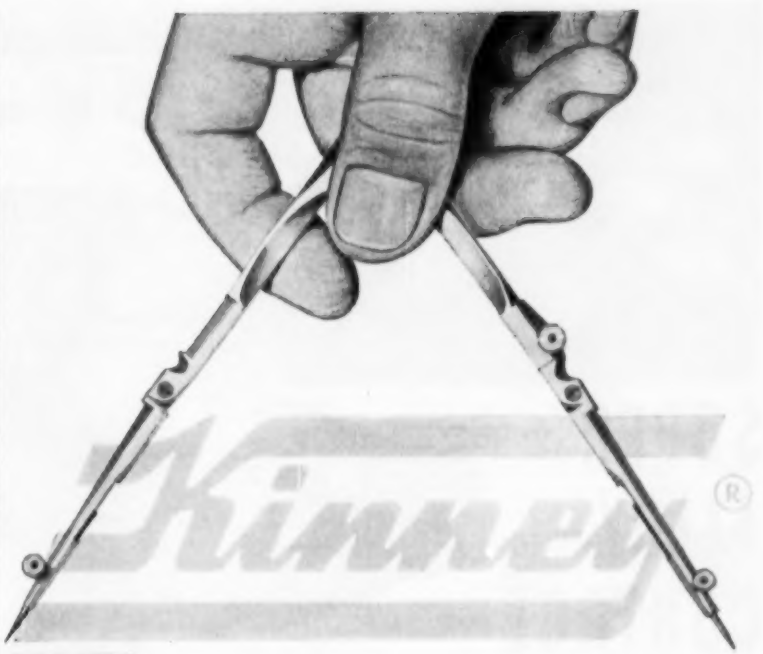
* "Turco" refers to compound No. 4182-A, which was supplied by Turco Products, Inc., Los Angeles. The manufacturers report that it is relatively noncorrosive to stainless steel, being free of chloride and fluoride ions. They state that it does contain sulfate ion as an impurity, which might cause corrosion if mixed with nitric acid wastes and stored in stainless-steel tanks. The company advises that by using higher-purity raw materials they can provide the same action in a product that is free of sulfate ion, which they designate Turco 4324. They advise also that 4324 solutions, after concentration by evaporation, will be free of anions other than phosphate.

bly require additional steps or special planning. Radioactive dust or metal chips might best be removed at the start with a vacuum cleaner.

Methods 4, 13, 15, and 16 for stainless steel represent approximations of procedures used to decontaminate stainless-steel vessels and other fixed process equipment at the chemical processing plant. Method 4 is seldom used due to its corrosive tendency. The other three procedures are preceded by Turco and 0.25*M* phosphoric acid. The treatment time for Turco and phosphoric acid was 20 min, as compared with 30 min for the other five procedures. The decontamination factors for Turco and phosphoric acid are several times those obtained with the citric, periodic, and nitric acid formulas. The possibility of substantially reducing the time required for decontamination of process equipment is thus suggested.

It was observed that the Turco solution dispersed the wax ring from the samples much more effectively than the phosphoric acid. The Turco solution was found to be alkaline in reaction, and it produced considerable foam when boiled. The wax, with its tendency to cover or retain contaminants, probably does not duplicate conditions found in lines and vessels. However, the possibility of organic solvents forming tars and gums should be considered. Some process-line valve parts showed residual contamination of 1.5 r/h after treatment by electrolysis and boiling in Turco; the activity was reduced to near background by boiling the parts in 0.25*M* phosphoric acid for 2 hr.

A sample of type 347 stainless steel having an area of 25 cm^2 was boiled for 24 hr in 0.25*M* phosphoric acid and lost 0.6 mg of weight. It was then boiled 10 min in concentrated nitric acid and lost 0.3 mg. If this represents a true measure of the metal lost it would be roughly equivalent to removing 0.002 mil from the surface. The true loss of metal is probably obscured by the formation of oxides or phosphates, which would add weight, and the true area is obviously greater than the apparent one. Since the sample was initially cleaned with nitric acid, it appears that the oxide coating was attacked by the phosphoric acid. It seems possible that a phosphate coating then formed that was insoluble in phosphoric acid but was attacked by nitric acid. The most recent tests were made with phosphoric acid; however, the preliminary results were promising and the phos-



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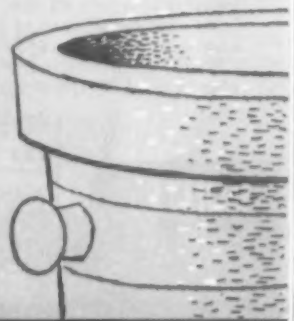

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phoric acid-nitric acid cycle is recommended for trial on difficult stainless-steel-decontamination problems.

Contamination mechanisms for stainless steel. Some speculation as to the reactions causing contamination on stainless steel seems justified. In procedure 16, over 1% of the measured activity remained after boiling with nitric acid. The retention of this residual contamination (which is generally the most difficult and costly to remove) cannot be satisfactorily explained as due to physical attraction. The most logical explanation seems to be that some of the fission products reacted with the surface of the disk to form compounds that were insoluble in nitric acid. Glasstone (*2*) points out that the oxide coating on passivated metals is impervious to oxygen and most anions; it may therefore be reasoned that it is impenetrable to the fission-product cations also. The reactions would thus be confined to the very thin oxide layer.

Mellor (*3*) reports the preparation of ferrites from copper, silver, nickel, cobalt, strontium, sodium and several other metals. Typical is zinc ferrite, $ZnO \cdot Fe_2O_3$, which occurs naturally in the mineral franklinite. Presumably, zinc nitrate (or the other metal nitrates) could react with ferric oxide to produce complex oxides of this nature. Mellor also reports that zinc ferrite can be dissolved in hot, concentrated sulfuric acid, and several of the other ferrites show a tendency to dissolve in organic acids. The large number of fission products present makes any analysis of the problem most complex, but the ferrite theory seems to offer a rough pattern for the trend shown in the stainless-steel list. It should be observed that chromium is also present in stainless steel, and that a comparable group of chromites might be formed.

* * *

The experimental work described in this report was done as a part-time, semi-official project; this work could not have been completed without the help of a number of individuals. Thanks are especially due to R. E. Hayden for a substantial amount of effort spent on procurement and other details, and to J. R. Bonnell, who recommended the Turco compound and discussed his experience in its use with the author.

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Assaying Compounds Containing H^3 and C^{14}

By GEORGE T. OKITA, JON J. KABARA, FLORENCE RICHARDSON and GEORGE V. LeROY

Argonne Cancer Research Hospital, The University of Chicago
Chicago, Illinois

The effectiveness of liquid scintillators for counting weak beta-emitting isotopes and the difference in the energies of the beta particles of tritium and carbon-14 make it possible to assay for both of these isotopes in a double-labeled compound. The maximum beta energy for H^3 is 0.018 Mev and for C^{14}, 0.155 Mev.

The instruments generally used are fast-coincidence circuits with one or more pulse-height discriminators. In some cases, the discriminator circuit is arranged so that one scaler monitors one portion of the energy spectrum, for which upper and lower discriminators may be set, and a second scaler monitors another portion.

In Fig. 1 is shown the spectral-energy curves for H^3 and C^{14} and the discriminator settings of a liquid-scintillation counter. The instrument used in this study was the Tri-Carb liquid scintillation spectrometer* with a two-channel pulse-height analyzer. Two channels for pulse-height discrimination are obtained by means of discriminator controls *A*, *B* and *C*. The output of each channel is fed into its respective scaler. All pulses of energy greater than *A* and less than *B* are monitored by the scaler connected with channel 1. Pulses with energies greater than *B* and less than *C* are registered by the scaler of channel 2;

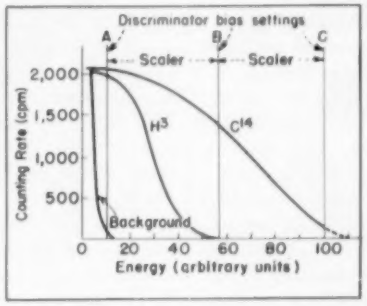

FIG. 1. Spectral-energy curves for tritium and C^{14} and discriminator bias settings of liquid-scintillation counter. Photomultiplier voltage was 1,130 volts. Curves plotted with discriminator ranges of 5 volts

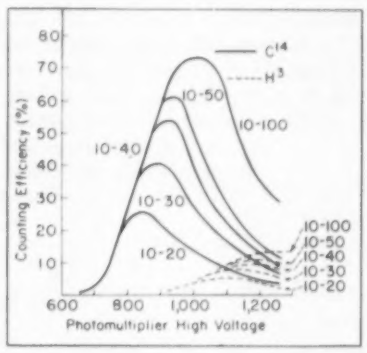

FIG. 2. Integral discriminator bias curves for tritium and C^{14}

* Tri-Carb Counter, Model 314, Packard Instrument Co., La Grange, Ill.

TABLE 1—Effect of Voltage and Discriminator Settings on Accuracy

Discriminator settings		*Standard error (%)**					
		1,130 volts		*1,200 volts*		*1,260 volts*	
First channel	*Second channel*	H^3	C^{14}	H^3	C^{14}	H^3	C^{14}
10–20	10–100	7.6	4.7	6.5	10.5	7.2	16.5
10–25	10–100	6.1	3.4	5.9	8.0	18.8	6.9
10–30	10–100	1.5	3.5	4.1	7.4	9.8	7.0

* Each figure represents standard error of four experiments.

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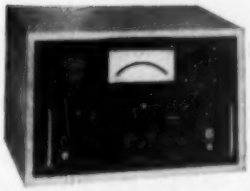

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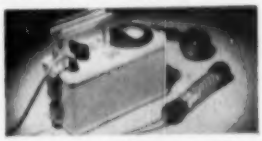

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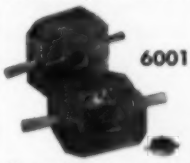

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this channel can also be set to monitor pulses with energies greater than A and less than C.

Optimum Settings

To differentiate between the two isotopes, as well as to obtain maximum reproducibility, it is necessary to select critically the proper pulse-height discriminator and photomultiplier voltage settings. To obtain the optimum settings, integral bias curves (Fig. 2) first were plotted for both isotopes. The optimum settings were determined by the point on the graph at which the highest "contrast" ratio (ratio of counts occurring in the second to the first channel) was obtained for both isotopes without an appreciable loss in counting efficiency in the first channel.

Discriminator control A for both channels was set at a pulse height of 10 volts, the point at which a good signal-to-noise ratio was obtained for the two isotopes. For the second channel, the "upper-gate" discriminator at C was set at the maximum pulse-height setting of 100 volts to eliminate large background pulses caused by local gamma radiation and cosmic rays. Therefore, the discriminator control requiring critical selection for the best "contrast" ratio is control B of the first channel. According to Fig. 2, the most desirable position for discriminator B is 20–30 volts. However, the 20-volt region may be less desirable due to appreciable sacrifice in counting efficiency in the first channel.

For selection of the photomultiplier voltage, it will be noted that any voltage lower than 1,000 volts will not be suitable for double-label counting since a "contrast" ratio of unity is obtained for tritium pulses below this value. Also, any voltage setting greater than 1,250 volts is undesirable because of the marked drop in C^{14} efficiency. Therefore, the voltage should be 1,100–1,200 volts. The drop in C^{14} efficiency can be attributed to over-pulsing of C^{14} pulses beyond the "upper-gate" discriminator when the photomultiplier voltage is increased.

For final selection of optimum settings, combinations of photomultiplier voltages in the range 1,100–1,250 volts and settings of discriminator bias control B of 20–30 were investigated using a mixed sample containing known amounts of tritium and C^{14}. The results of such determinations are shown in Table 1. The greatest accuracy was obtained with a 1,130-volt photomultiplier setting and a discriminator range of 10–30 for channel 1 and 10–100 for channel 2.

The photomultiplier voltage and discriminator settings will vary from instrument to instrument depending upon the condition of the phototube, electronic circuitry, type of solvent system employed, etc. It should be emphasized that it will be necessary to check the integral bias curves from time to time to note any shift.

Assay Methods

We have used three methods for counting the two isotopes in the same sample to determine which gives the best accuracy. They are the screening, the simultaneous-equation, and the discriminator-ratio methods. As applied to an instrument of the Tri-Carb type, descriptions of the three procedures follow.

Screening method. This procedure requires counting both isotopes simultaneously at a given discriminator and photomultiplier voltage. Then the C^{14} is counted alone at a lower photomultiplier voltage to screen out all tritium pulses. It is not possible to screen out all tritium pulses by holding the voltage constant and setting discriminator control B high.

The radioactivity for each isotope can be calculated by applying the following formulas, where $C = C^{14}$ dpm, $H = H^3$ dpm, N_1 = net cpm at low voltage, N_2 = net cpm at higher voltage, $h_1 = H^3$ efficiency at low voltage, $h_2 = H^3$ efficiency at higher voltage, $c_1 = C^{14}$ efficiency at low voltage, and $c_2 = C^{14}$ efficiency at higher voltage. Thus $N_1 = Hh_1 + Cc_1$. If $h_1 = 0$ and $c_1 > 0$, then

$$C = N_1/c_1$$

Also $N_2 = Hh_2 + Cc_2$. If h_2 and $c_2 > 0$, then

$$H = \frac{N_2 - Cc_2}{h_2}$$

Simultaneous equation method. In this method, the photomultiplier voltage was held constant. Discriminator control B was set so that an appreciable amount of both tritium and C^{14} pulses could be detected in both channels. The discriminator range for the first channel was from A to B and for the second channel from A to C.

The activity of each isotope was calculated by applying the following formulas that were derived by setting up simultaneous equations to solve for the two unknowns. N_1 = net cpm of channel 1, N_2 = net cpm of channel 2, $h_1 = H^3$ efficiency of channel 1, $h_2 = H^3$ efficiency of channel 2, $c_1 = C^{14}$ efficiency of channel 1, and $c_2 = C^{14}$ efficiency of channel 2.

$$C^{14}\ \text{dpm} = \frac{N_1h_2 - N_2h_1}{c_1h_2 - c_2h_1}$$

$$H^3\ \text{dpm} = \frac{N_1c_2 - N_2c_1}{h_1c_2 - h_2c_1}$$

Discriminator-ratio method. In the discriminator-ratio method, the photomultiplier-voltage and discriminator control settings are similar to those used in the simultaneous-equation method. The discriminator-ratio procedure circumvents the use of efficiency factors, which are only as accurate as the original assay of the tritium* and C^{14}† standards.

This method uses the ratio of channel 2 cpm to channel 1 cpm, which are obtained from standard samples of each of the isotopes. An equation is then derived that utilizes the discriminator ratio for each isotope instead of the counting efficiency as used in the two previous methods.

To derive the equations for calculating the activities for each of the isotopes, N_1 = net cpm of channel 1, N_2 = net cpm of channel 2, H_1 = net cpm of H^3 in channel 1, H_2 = net cpm of H^3 in channel 2, C_1 = net cpm of C^{14} in channel 1, C_2 = net cpm of C^{14} in channel 2, $a = H_2/H_1$ and $b = C_2/C_1$. Since

$$N_1 = H_1 + C_1$$

and

$$N_2 = H_2 + C_2$$

Then one can derive

$$H_1 = \frac{bN_1 - N_2}{b - a} \qquad (1)$$

and

$$H^3\,\text{dpm} = \frac{H_1}{\text{channel 1 } H^3 \text{ efficiency factor}} \qquad (2)$$

One can also derive

$$C_2 = \frac{b(N_2 - aN_1)}{b - a} \qquad (3)$$

$$C^{14}\ \text{dpm} = \frac{C_2}{\text{channel 2 } C^{14} \text{ efficiency factor}} \qquad (4)$$

The following is a sample calculation

* Tritium-labeled cholesterol used as the tritium standard was assayed by the method of Wilzbach, *et al.* (*1*).

† Cholesterol-4-C^{14} used as the C^{14} standard was assayed against Na_2CO_3-C^{14} obtained from the National Bureau of Standards. The method of assay was the wet-combustion and ion-chamber-vibrating-reed-electrometer method of Brownell and Lockhart (*2*).

of a double-labeled compound using the discriminator-ratio method. The sample contained a known activity of 10,800 dpm of H^3 and 2,928 dpm of C^{14}. Setting the photomultiplier voltage at 1,130 volts and the discriminator at 10–30 volts for the first channel and 10–100 volts for the second channel, the following figures were obtained: $N_1 = 561$ cpm, $N_2 = 3{,}759$ cpm, $a = 3.30$, $b = 16.8$, channel 1 tritium efficiency factor = 0.0382 and channel 2 C^{14} efficiency factor = 0.816. Therefore, calculated by means of Eqs. 1 and 2, the tritium activity is

$$\frac{bN_1 - N_2}{b - a} = \frac{16.8 \times 561 - 3{,}759}{16.8 - 3.30}$$

$$= 418 \text{ cpm of } H^3$$

$$H^3 \text{ dpm} = \frac{418}{0.0382} = 10{,}942$$

Similarly, by means of Eqs. 3 and 4, the C^{14} activity is 2,886 dpm.

Reliability of Methods

To determine which method was the most reliable, twelve samples containing varying amounts and ratios (H^3 dpm/C^{14} dpm was 1–10) were assayed by the three methods. The standard error of the assay was used as a criterion of accuracy (S.E. = $\sqrt{\Sigma d^2/n(n-1)}$, where d = per cent deviation from theoretical). The data is tabulated in Table 2.

The discriminator-ratio method proved to be the most reliable of the three methods. The standard error for the tritium assay was 3.7% and for C^{14}, 4.6%. The larger error obtained by the screening method may be attributed to the difficulty in eliminating all tritium pulses when the discriminator bias and photomultiplier voltage are set to assay only C^{14}.

The lower reliability of the simultaneous-equation method may be attributed to the fact that it requires using the counting efficiencies for both iso-

TABLE 2—Accuracy of Methods

Method of assay	*Standard error (%)** H^3	C^{14}
Screening	7.6	6.7
Simultaneous equation	4.7	8.4
Discriminator ratio	3.7	4.6

* Each figure represents standard error of 12 samples containing varying amounts of tritium and carbon-14 activity.

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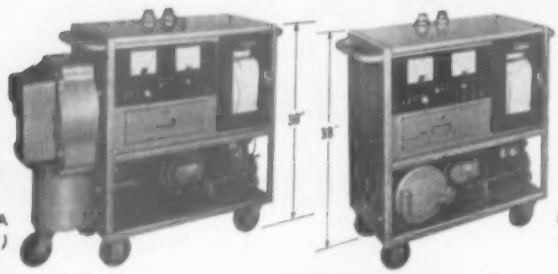

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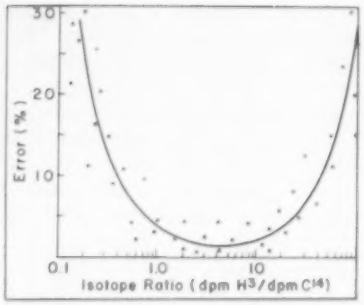

FIG. 3. Effect of isotope-ratio variation on accuracy of assay

topes six times. Since the counting efficiencies are only as accurate as the assay of the original isotope standard, this may introduce an appreciable error if there was a slight error in the original assay.

Other Factors

For the most reliable results in counting double-labeled compounds, it is necessary to consider the scintillator-solvent system, the isotope-ratio and radioactivity level of the sample.

Change in scintillator-solvent system. Each scintillator-solvent system has its own characteristic integral discriminator bias curves for tritium and C^{14}. It is therefore essential that the discriminator and photomultiplier-voltage settings be redetermined whenever the composition of a system is changed.

Isotope ratio in sample. Effect of variation in the isotope ratio on assay reliability was also determined. As can be seen from Fig. 3, for accurate assays H^3 dpm/C^{14} dpm should be 1–15.

Minimum radioactivity level. It is of course essential that all sample activities be sufficiently greater than the background rate. For routine radioassay, it was found that for a background counting rate of 40 and 60 cpm for channel 1 and channel 2 discriminators, respectively, and for a counting period of less than 30 min, the sample-to-background counting-rate ratio should be no less than three.

* * *

The authors wish to acknowledge the technical assistance of Messrs. Roger Pating and Roh-Leong Tan. We are indebted to Mr. Leo Slatery of the Packard Instrument Co. for suggesting the discriminator-ratio method.

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2. G. L. Brownell, H. S. Lockhart, NUCLEONICS **10**, No. 2, 26 (1952)

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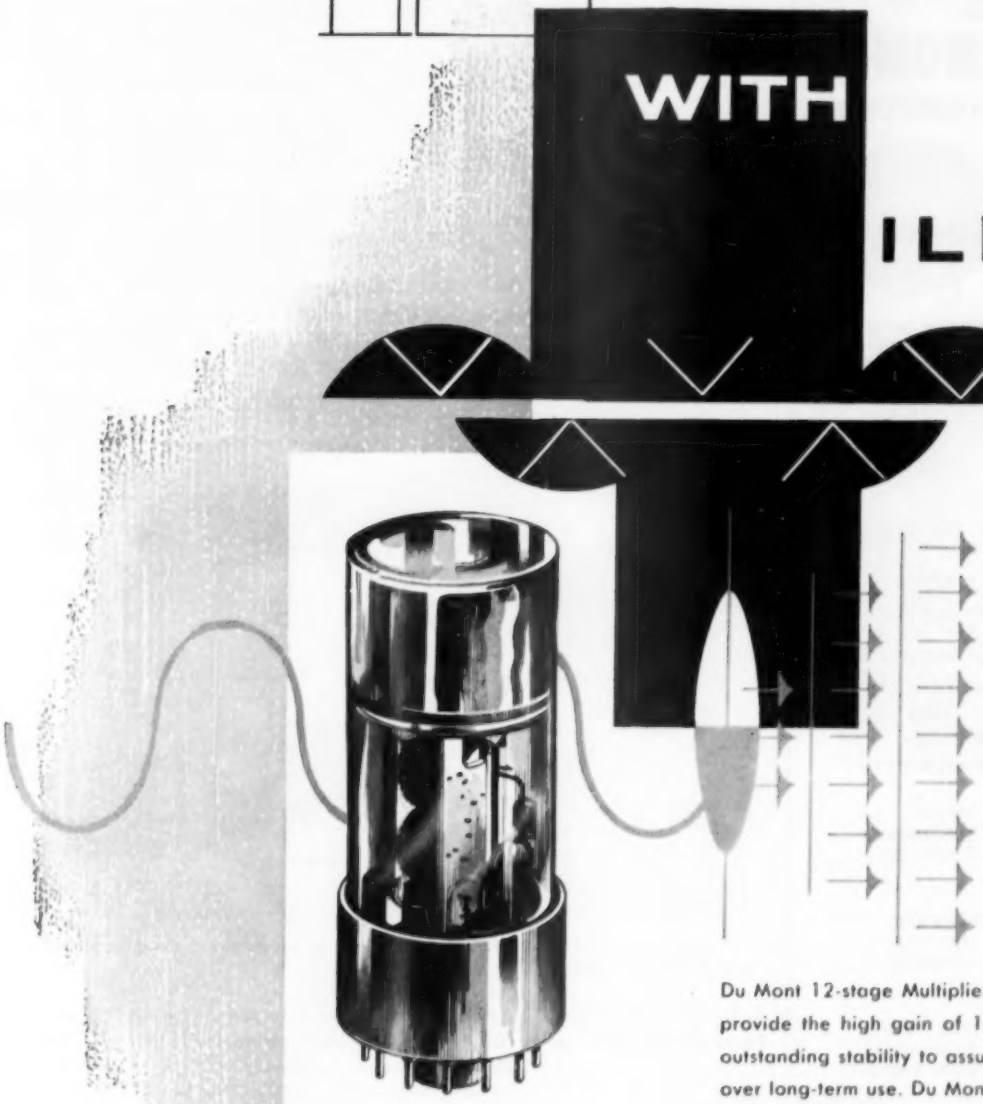

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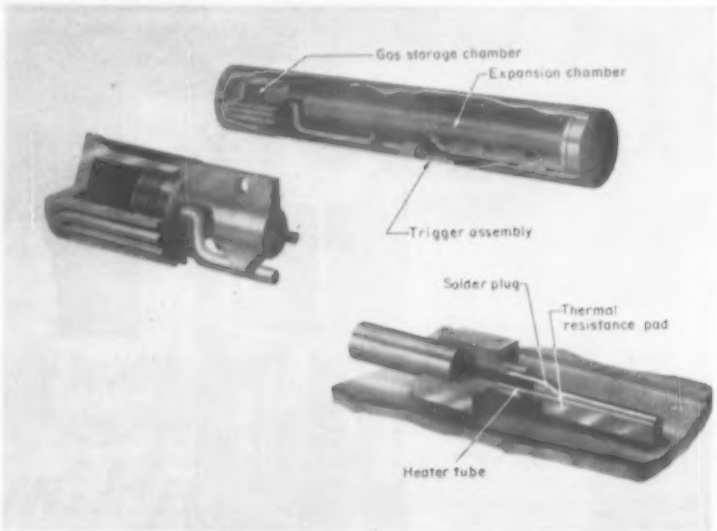
NUCLEAR ENGINEERING NOTES

Reactor Fuse: New Safety Element

A reactor can now be scrammed by a device that is independent of electrical or mechanical power source. The reactor fuse, being developed by Atomics International, is located in the core. If the neutron flux level becomes excessive, or the reactor coolant is lost, a solder plug melts and boron trifluoride (enriched in B^{10}) expands from a small upper chamber into a large lower one (see cut). The gas, in expanding into the lower chamber, increases in neutron-absorbing ability by a factor of 3–5 because of decreased self-shielding.

Here is how it works: The tube containing the solder plug is made of uranium oxide-copper compact (~25 wt% UO_2) sintered and coined to a high density. The UO_2 is enriched in U^{235}. Fission heat from the uranium is removed through a stainless-steel resistance pad that is bonded to the aluminum container wall. At normal flux levels, the temperature difference across the resistance pad maintains the tin-silver eutectic plug somewhat below its melting point. If the flux level becomes excessive or the coolant is lost, the fission heat raises the temperature of the plug to its failure point so that it is ejected by the high-pressure BF_3. High speed of gas expansion makes the reactor fuse 90% effective within 150 msec after the solder plug melts. The burned out fuse must be replaced, of course.



CUTAWAY VIEWS of reactor fuse showing details of gas storage chamber (upper left) and of trigger assembly (lower right)

Subcritical Reactors New Power Source?

A scheme that would allow any reactor to be operated subcritically has been proposed by Lyle B. Borst, Head of the Physics Department, New York University. A device called the Convergatron would maintain the neutron flux in the subcritical reactor at a level determined by the neutron contribution from a small external neutron source. The reactor could then be directly controlled by adjusting the external source strength and could be turned off simply by removing the source.

At the moment the design of the Convergatron is still in the conceptual stage. However Borst is optimistic about the solution of the engineering problems that would arise in incorporating his device into a reactor system. He feels that his proposed scheme would lead eventually to an economic advantage for commercial power reactors by eliminating expensive safety features such as elaborate control systems and containment vessels.

Convergatron principle. The Convergatron is based on the principle that a subcritical system behaves as a neutron amplifier. If n neutrons are introduced into the system from an external source the system will yield a total of $n/1\text{-}k$ neutrons where k is the reactivity. Thus a subcritical assembly is like an amplifier with a gain of $1/1\text{-}k$. The Convergatron consists of a string of subcritical systems arranged in such a way that the neutrons produced in one system act as source neu-

trons for the next one and so on. The over-all amplification of the series is then the product of the amplification factors of the individual stages (multiplied, in addition, by an efficiency factor to take into account unavoidable losses between stages*). Thus for eight stages each with $k = 0.9$ the over-all amplification factor would be close to 10^8.

Borst's idea is to use an external neutron source as the input for a Convergatron and to feed the greatly amplified output into an ordinary reactor that is kept subcritical. The reactor will then operate at a neutron production rate that is some multiple of the Convergatron output. Since this output in turn is strictly determined by the Convergatron input, the reactor neutron flux level will be related directly to the neutron source strength at the input.

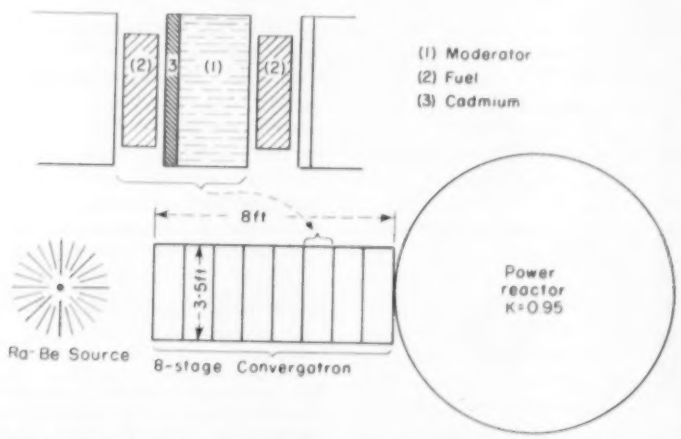
A stage consists of three regions: (1) a region of pure neutron moderator such as water or graphite (2) a region containing fuel, uranium or plutonium and (3) a thermal neutron barrier such as cadmium, arranged in cyclic order. Fast neutrons from fission of uranium in the fuel zone penetrate the cadmium barrier. They are slowed down by the moderator to become thermal neutrons which activate the next fuel zone. Those fast neutrons starting in the reverse direction are slowed down by the moderator until they can no longer penetrate the cadmium.

Each fuel zone receives neutrons from the preceeding stage. Each succeeding stage can therefore operate at a higher power while still depending upon the preceeding stage for its excitation.

Preliminary design. Although a detailed engineering design has not yet been worked out, Borst has made preliminary estimates that indicate a reasonable physical size for the Convergatron system. For a reactor with a flux in the neighborhood of 10^{15} n/cm² sec, he believes that eight amplifying stages would be required if a strong Ra-Be source were used at the input or six stages if a Van de Graaff were used. An amplifying stage might be in the shape of a disk, 1 ft thick and 3–5 ft in diameter. Eight amplifying stages stacked together would then form a cylinder 8 ft long. The total fuel inventory would amount to 16 kg.

The primary advantage of the system would be its inherent safety. Since the reactor power could be completely controlled by the neutron source, control rods capable of making the reactor critical would be absent from the design. Thus all the parts of the system could be designed to be permanently subcritical and hence the danger of the operator losing control of the chain reaction is removed.

One of the engineering obstacles is that the power produced in the last stage of the Convergatron is a significant fraction of the total power. If, for instance, the reactor proper has a $k = 0.95$ then the last stage must supply 5% of the neutrons produced in the reactor. If the total reactor power is 300 Mw then a power of the order of 30 Mw is implied for the last stage. An engineering design is needed that will permit the removal of this heat without adding to the complication and cost of the reactor system.



SUBCRITICAL REACTOR operation is made possible by addition of Convergatron unit. Neutron production from Ra-Be source is amplified to provide extra neutrons needed to sustain flux level in subcritical power reactor

* The value of this efficiency factor and therefore the actual amplification per stage is still uncertain. Borst plans to make more accurate calculations of amplification using transport theory and Monte Carlo techniques.

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Magnetic Jack—a New Control Drive Mechanism

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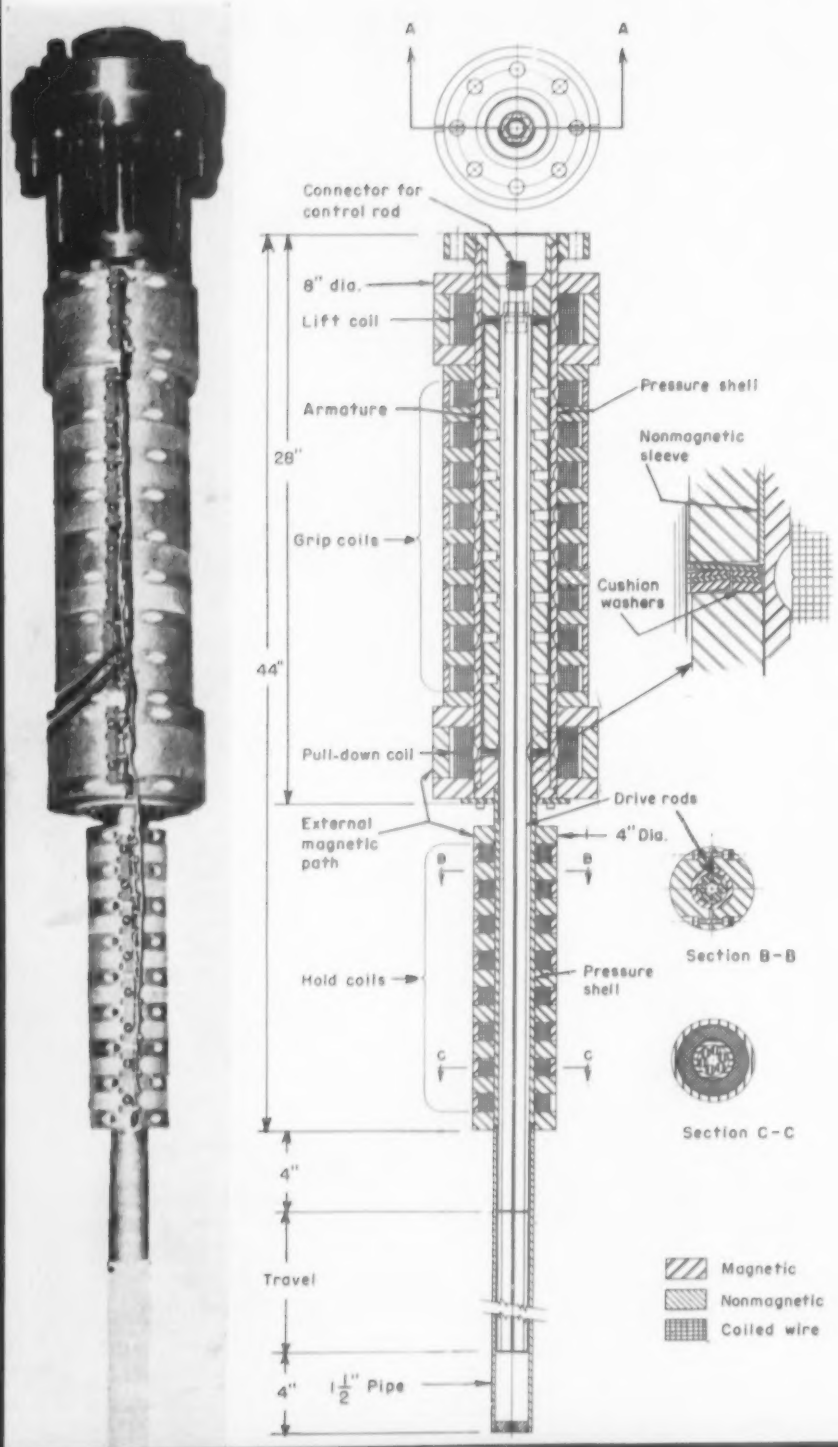

We needed a control drive that would operate reliably in a water cooled reactor. The first drive we designed had a ball-bearing screw driven by a canned-rotor motor and a roller scram latch actuated by an electromagnet. This required many precision parts, some of which were of cobalt alloy. The final design was expensive and unreliable. It became apparent that it is not practical to operate a complicated, precision mechanism in water.

Work was then begun on the design of an electromagnetic drive employing magnetically actuated gripper mechanisms. This device gave somewhat encouraging results, but showed that even simple mechanical mechanisms are not practical if they must carry a load in water.

A method of gripping magnetically, without the use of a mechanical gripper mechanism, has been evolved. Three models of this type have proved very satisfactory in extensive testing. We call this type of mechanism the magnetic jack.

Functional Description

Essentially, the magnetic jack consists of a pressure shell, external stationary magnet coils, and one internal moving part that imparts linear motion to a bundle of drive rods. The control rod itself is attached to the drive rods. The rod is moved in precise steps by switching the four sets of coils on and off in a certain sequence. The motion is reversed simply by reversing the sequence. The rod is held stationary by stopping the switching. The rod is

FIG. 1. Magnetic jack for installation below reactor vessel. System of coils can be energized in turn to jack control rod magnetically. Neutron-absorbing portion of control rod is at top end in operation. Pushing rod up through reactor core with magnetic jack removes neutron absorber from core. Turning off power to jack allows neutron absorber to fall back into core by gravity

scrammed when the current to the coils is turned off or fails.

The section of the control-rod assembly that passes thru the magnetic jack must be magnetic and able to expand radially. A cluster of six round rods of magnetic (nonpermanent!) stainless steel shown in Fig. 1 is one arrangement that satisfies this requirement. The cluster of rods fits loosely in the center of the jack, except when the current to the hold or grip coils is turned on. The small hold coils pull the rods out radially and hold them to the wall of the pressure shell by magnetic attraction and friction. The grip coils pull the rods to the wall of the armature so that the rods move if the armature moves.

The armature has an axial clearance that determines the length of the step of this step-motion, linear motor. The practical range of step length is 0.02–0.20 in. The large coil at the top end of the armature lifts it up, and the large coil at the bottom end of the armature pulls it down—if it doesn't drop by gravity.

The armature and the end coil constitute a short-stroke solenoid which gives a much stronger pull than an ordinary long-stroke solenoid for the same amount of power because of the short magnetic flux gap. For the proportions shown in Fig. 1 and for a coefficient of friction of 0.15, the combined holding force of eight holding magnetic fields equals the force of the lift magnet.

Figure 2 gives the sequence for switching the four sets of coils off and on to move the rod in steps.

Advantages

A control-rod drive must perform the following functions:* (a) move the rod in either direction at a controlled rate; (b) hold the rod at a fixed position; (c) indicate the position of the rod; (d) be able to release the rod quickly under any condition. At the same time, the drive must accomplish these functions precisely, without failure, over a long period of time with or without lubrication; operate in a small space with little cooling; and be economical to build and maintain.

Tests show that the magnetic jack meets these requirements adequately and has the following advantages:

Reliable. The magnetic-jack has never failed mechanically during extended tests of the three models that have been built. This is important, since the interior of the mechanism is sealed off, and the jack is located near the reactor. The jack will hold the rod at any given position as long as the grip and/or hold coils are supplied with current.

Fail safe. All of the failures encountered during the testing of the three magnetic jacks were electrical. The failures either caused the rod to drop back or did not step it up; they never prevented the rod from dropping.

Rugged. The rod can be driven against the stops at the ends of its travel and the drive stalled. The rod can be released and then regripped while dropping. Overloading, such as from a mechanical shock, would cause some slippage, but no damage. If a control rod becomes stuck, it can be driven down with a force equal to the lifting capacity.

Long lived. Although there is some wear during operation, the parts can

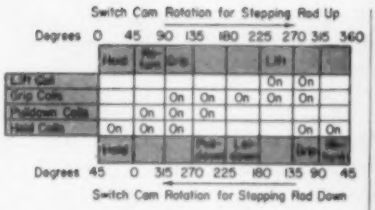
Switch Cam Rotation for Stepping Rod Up
Degrees 0 45 90 135 180 225 270 315 360
Lift Coil
Grip Coils
Pulldown Coils
Hold Coils
Degrees 45 0 315 270 225 180 135 90 45
Switch Cam Rotation for Stepping Rod Down

FIG. 2. Starting with rod held stationary (only hold coils on), to step rod up, pull-down coil is energized to ensure that armature is down; grip coils are energized to adhere drive rods to armature, hold coils and the pull-down coil are de-energized, separately, with drive rods held to armature but free in jack; lift coil is energized to lift armature, thus moving rod up one step; hold coils are energized to hold rod stationary; grip coils and lift coil are de-energized to complete cycle. To reverse motion of rod, sequence is reversed. One simple cam actuates all four sequencing switches

* J. M. Harrer, NUCLEONICS **13**, No. 6, 48 (1955).

Use in Other Reactors

Several variations of the design described here are available for different applications. One, for unpressurized research reactors, is a low cost magnetic jack which uses standard magnetic-relay parts. Another has the added feature of requiring very little head room. It consists of a cable drum, an electric clutch, an electric brake, a rotary solenoid, and a helical potentiometer for position indication—all on one shaft. This rotary magnet jack operates similarly to the linear magnetic jacks, except that the return spring on the rotary solenoid replaces the pulldown system.

It is noteworthy that variations in length of rod travel for different application can be accommodated merely by installing a longer drive rod and pipe on the same drive mechanism.

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take a lot of wear before malfunction occurs. Unplated stainless-steel rods seem to be adequate, however, chrome plated rods would last much longer.

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Sealless. Since power and control is transmitted to the drive by magnetism through the pressure shell, no shaft seals or pressurized electrical connections are required.

Inexpensive. The cost of building the magnetic jack and the power supply was $2,900. The mechanical parts could be made in an ordinary machine shop, since the parts are simple and do not need to be precise. No expensive materials are required.

Limitations

The test results have also indicated the following limitations:

Limited position indication. Since there are no shafts or rods coming out through the pressure shell, the position of the rod must be determined indirectly.

Limited position adjustment. The magnetic jack moves the rod in steps or increments, and the steps cannot be very small unless a very slow rod travel is desired.

Limited shock resistance. With the parts free of oil and with both hold and grip coils on, the holding capacity is about 1,400 lb. Assuming a 60-lb control rod and a 40-lb drive rod, the jack can resist a 14*g* shock while holding, and a 6.5*g* shock while moving the rod. Greater shock loads would cause the rods to slip slightly.

Limited lifting capacity. The jack has a maximum lifting capacity of about 800 lb and a rated lifting capacity of 600 lb.

Limited speed. Although the rod can be dropped rapidly, the maximum speed with which the rod can be driven (in either direction) is limited to about 6 in./min.

Operating Characteristics

To determine its operating characteristics, the jack was mounted below a 20-ft-high pressure vessel that simulated a boiling-water reactor. The top of the vessel contained 600-psig saturated steam. The rest of the vessel, including the jack, contained 600-psig water at 488° F near the top and 80° F near the jack. The lower 4 ft of the vessel, which simulated the thimble, was uninsulated. The thimble and the jack were cooled by ambient (75° F) air.

The total load imposed on the jack was 220 lb and the speed at which the load was moved was 30 steps/min. The power used for lifting the load was 670 watts and the power used for holding it was 160 watts.

The jack was tested for a period of 5 months, of which 3,150 hours were at 600 psig and the remaining 450 hours at atmospheric pressure. The total step-motion travel was 18,300 ft. The total number of fast-return strokes (scrams) was 2,200.

The following are the results of the operating trials.

Power requirement. Figure 3 shows the minimum power required to lift, grip and hold various loads. Usual operating power is about three times the minimum required. Thus, the average input power for lifting was 1075 watts at 86 volts d-c.

Lifting speed. The magnetic jack operates best at speeds of less than 45 steps/min. Higher rates caused a decrease in step length, thus automatically limiting the speed of rod withdrawal to about 6 in./min. Most of the testing was done at a speed of 2.4 in./min.

Step length. When the jack mechanism was new, the coefficient of friction was about 0.15. Operating the mechanism with water in it for several days, increases the coefficient of friction by a factor of about three. This is probably due to the removal of the last trace of lubricant and/or the formation of iron oxide. When the jack was new, the step length was 0.075 in. up and 0.083 in. down. After several days operation

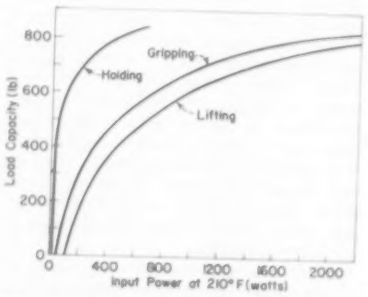

FIG. 3. Power requirement for magnetic jack. Data obtained on test stand with jack containing unpressurized water

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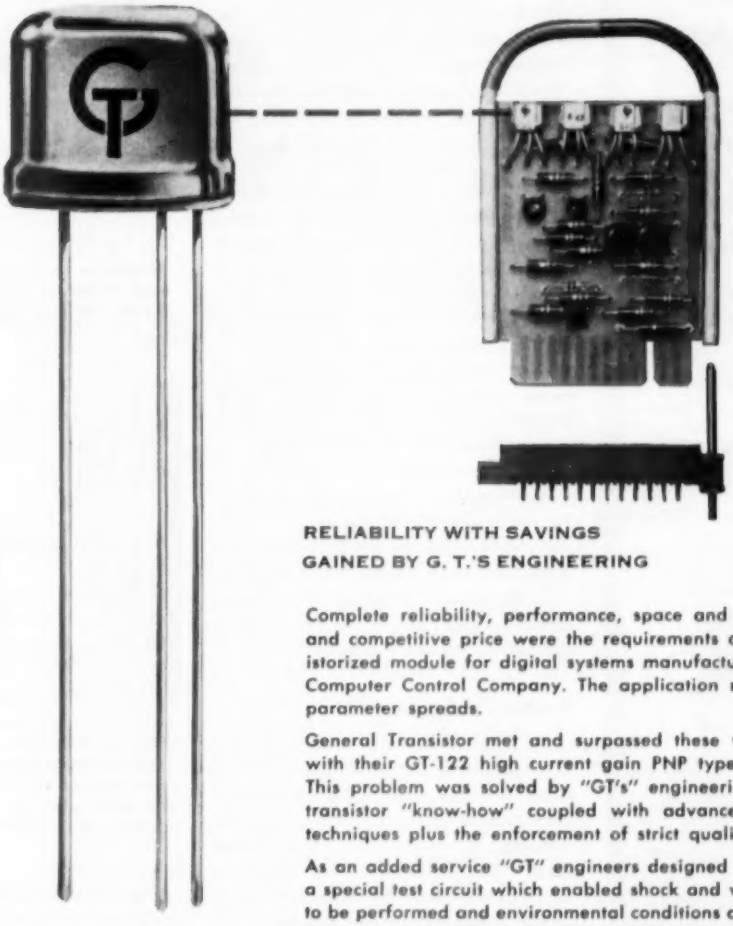

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with water in it, the step length was 0.080 in. up and 0.082 in. down.

Heating. After running continuously, the temperature of the hottest coil was 260° F and the hottest part of the outside surface of the jack was 195° F. After holding continuously, the temperature of the hottest coil was 210° F.

Fast return stroke. The delay time depends upon the method of shutting off the power and the amount of excess holding power. With ordinary breaking of the d-c power to the coils, and with twice the required holding force, the delay time was about 20 msec. The fast-return action of the multiple-rod type magnetic jack has never failed. The release characteristics are the same after holding several days as they are after holding only a few seconds.

Wear. Wear ranged from 0.009 in. on the drive rods to negligible wear on all other components. The cushion washers were pitted, but still serviceable. Since the wear was so slight, it is safe assumption that 5 months represents a small fraction of the useful life of the magnetic jack.

Dirt accumulation. The internal parts became covered with a black powder, probably oxidized wear particles. There was no dirt buildup on contacting surfaces.

Malfunction. There were no malfunctions of the internal parts. Externally, the flange gasket leaked on two occasions, and a lead wire to one of the coils shorted. There were a few breakdowns of the power supply. These difficulties are being remedied.

Position Indication

Position indicators used with the magnetic jack are of the electromagnetic, stationary-coil type. Basically, they consist of a coil that fits over the nonmagnetic portion of the pressure shell. The rods, which are magnetic and long enough so that a part is always in the coil, increase the impedance of the coil as they move into it. Measuring the impedance gives the position of the rod.

Figure 4 shows the wiring diagram of the basic system. The coil is made of constantan wire so that changes of temperature affect its resistance only slightly.

To calibrate the system, a shorter standard coil is added. This coil is located so that the rod always extends through and beyond the coil so that it is effected by the magnetic properties of the rod but not by the movement of the rod. Thus, the system can be calibrated with the rod in any position. Figure 5 shows the wiring diagram for this two-coil system. Calibration is accomplished by switching from the position coil to the standard coil and adjusting the rheostat to make the meter read a given value.

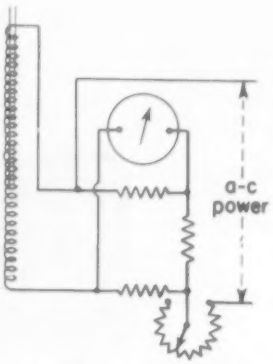

FIG. 4. One-coil position indicator. To provide linear indication of position, each resistor of bridge circuit has same d-c resistance as coil

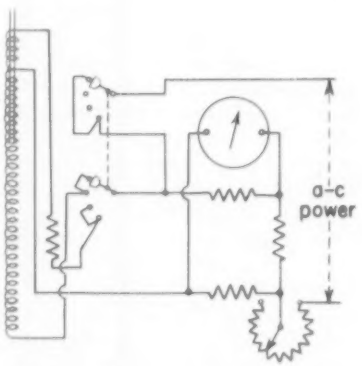

FIG. 5. Two-coil position indicator. Calibration is accomplished by switching from position coil to smaller standard coil and adjusting rheostat to make meter read specified value

The preceding arrangement compensates for all variables except variation in temperature along the length of the rod. To minimize this error, the multiple-coil system shown in Fig. 6 can be used. The temperature-variation error is reduced in this system because a shorter length of rod affects the coils. As the end of the rod moves from one set of coils to the next, the inactive coils are switched out of the circuit, and the active coils are switched in by a selector switch. The selector switch setting roughly indicates the rod position. The meter gives the fine reading. To provide a fairly linear

meter reading, two adjacent coils can be used as one coil, but switching is provided from one single coil to the next to give an overlapping effect.

One of the multiple-coil position indicators that has been tested uses a coil length of 3 in. It is accurate to within 0.1 in. over the full travel and shows movement with an accuracy of 1%. Greater accuracy can be obtained by using a greater number of shorter coils, but the disadvantage of this is the necessity of using more wires from the control-rod drive to the control room. The operation of the selector switch and the rheostat can be made automatic, but this has the disadvantage of requiring a more complicated receiver.

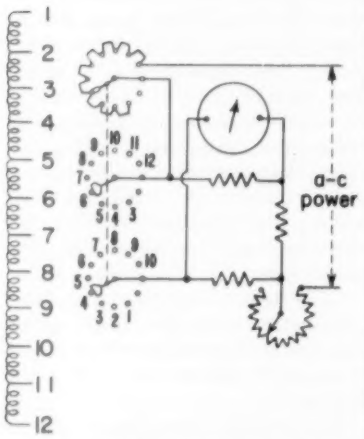

FIG. 6. Multiple-coil position indicator. To read position, selector switch is turned toward side where meter points until meter reads in active zone. Meter reading is added to reading of switch. To calibrate, selector switch is turned left two notches from notch which makes meter read in the active zone. Rheostat is adjusted to make meter read infinity

To eliminate the need for a selector switch, a separate meter for each pair of overlapping coils can be used. In this case, the position would be read directly on the meter that reads in the active zone, and the system would be calibrated by making one of the other meters read infinity.

If close measurement of the rod position is not essential, the meters mentioned above can be replaced by lights and the calibration rheostat eliminated. Such a system, using a coil length of 2 in., has been tested. It was easy to estimate the rod position to within 1 in., and step movements of 0.08 in. caused a visible change in light intensity.

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3/8	
1/2 & Up	

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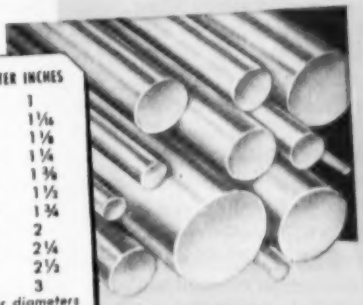

DIAMETER INCHES	
1/4	1
5/16	1 1/16
3/8	1 1/8
7/16	1 1/4
1/2	1 3/8
9/16	1 1/2
5/8	1 3/4
3/4	2
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2 1/2	1 1/2
3	1 3/4

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Redundancy, Coincidence and Built-In Testing Lead to . . .

Reliable Reactor Protection

By E. SIDDALL
Atomic Energy of Canada Limited, Chalk River, Ontario

Canada has, to a considerable extent, concentrated on heavy-water-moderated reactors using natural or only slightly enriched uranium as fuel. This combination results in reactors that are highly rated (in terms of power density) to make economic use of the expensive heavy water. A high rating implies a high flux. A high flux results in a very considerable buildup of Xe^{135} after shutdown. To start up after an unnecessary scram caused by an electrical failure, a fairly rapid rate of increase in reactivity must be possible in order to overtake the buildup of xenon poison.

This characteristic, coupled with their quite small temperature coefficients, make Canadian reactors necessarily much more dependent on their control systems than are some other reactor types. We have, therefore, explored very thoroughly the possibility of developing control systems that are adequately safe in themselves, rather than rely on various features that reduce the seriousness of the consequences of a control-system failure.

Early efforts to meet this requirement resulted in the use of a large number of protective devices that were designed to be "fail-safe" in as many respects as possible. The inevitable result was an excessive number of cases in which the reactor was shut down because of minor trouble in the electrical circuits and instruments.

These early efforts led to the use of protective channels of the type shown in Fig. 1. In this scheme, the principal trip instruments are duplicated and connected to two different trip lines so that both communication channels, from the instruments of a pair to the final circuit, must be disabled simultaneously before the system as a whole is disabled.

There are, however, some serious shortcomings in protective systems of the type shown in Fig. 1:

(a) The actual condition of the system at a given time is not at all obvious. Thus, the person responsible for operation will usually be dependent on the advice of a special instrument technician.

(b) It is not possible to test the entire system in operation. Some of the individual instruments can be tested, but this necessitates disabling and disconnecting the instrument, and there is a danger that it may not be restored to service correctly. Also, since a power reactor will be in operation for long periods, the probability of the system having deteriorated since the last test can be appreciable.

(c) Many systems used at the moment have obvious "bottlenecks" at which a relatively simple deliberate action can disable the protection. The points X and Y in Fig. 1 are an example. If these are energized by a separate voltage source, or connected directly to the battery, the whole system is disabled without the fact being obvious.

This situation gave rise to serious concern about the reliability of reactor protective systems. Since equipment improvements seemed insufficient to meet the problem in the immediate future, new philosophies of control-system design had to be evolved.

Performance of Systems with Redundancy*

No. of subsystems	*No. of signals for complete trip*	*Probability that system is disabled by unsafe failure*	*Typical value* (yr^{-1})	*Probability of accident because system so disabled*	*Typical value* (yr^{-1})	*Frequency of shutdown because of safe failures, tests & overhauls*	*Typical value* (yr^{-1})
1	1	$n_u t_u$	2.3×10^{-2}	$n_a n_u t_u$	4.6×10^{-2}	n_s	20
2	1	$(n_u t_u)^2$	5.3×10^{-4}	$n_a (n_u t_u)^2$	1.1×10^{-2}	$2n_s$	40
2	2	$2n_u t_v$	4.6×10^{-3}	$2n_a n_u t_v$	9.2×10^{-4}	$2n_s(n_s t_s + n_t t_t + n_o t_o)$	0.14
3	2	$3(n_u t_v)^2$	1.6×10^{-5}	$3n_a (n_u t_v)^2$	3.2×10^{-5}	$6n_s($ " " ")	0.42
4	2	$4(n_u t_v)^3$	4.9×10^{-8}	$4n_a (n_u t_v)^3$	9.8×10^{-8}	$12n_s($ " " ")	0.8
3 + 3	2 or 2	$9(n_u t_v)^4$	2.5×10^{-10}	$9n_a (n_u t_v)^4$	5.0×10^{-10}	$12n_s($ " " ")	0.8

* **Definitions of symbols** (assumed values in parenthesis): n_u = frequency of unsafe failures in each subsystem (2/yr), n_s = safe failures (20/yr), n_t = test trips (500/yr), n_o = trips due to overhaul (50/yr), n_a = occurrence of emergencies requiring operation of system (2/yr), t_u = average duration of unsafe failures in each subsystem without over-all test (100 hr), t_v = with overall test (10 hr), t_s = safe failures (0.1 hr), t_t = test trips (10 sec), t_o = trips due to overhaul (0.5 hr).

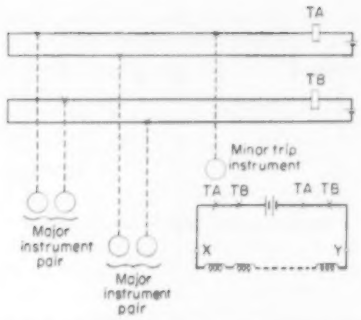


FIG. 1. Double protective system employs principle of redundancy—warning from either instrument of pair trips circuit. Weakest point in logic of this scheme is single circuit that energizes holdup magnet

Redundancy and Coincidence

The new philosophy was partly provided in the form of the principles of redundancy and coincidence. The principle of redundancy provides "safety in numbers" in reactor protective systems. Obviously, only one instrument is necessary to monitor a given transient phenomenon. If only one instrument is provided, however, and that instrument fails, an unsafe condition may go undetected. If, on the other hand, two or more instruments are provided to monitor the same phenomenon, the maloperation of one instrument will not prevent an unsafe condition from being detected. Thus, the safety of the reactor system is increased by each extra, or redundant, instrument.

The double system of Fig. 1 is an example of redundancy. Each instrument or subsystem can show a safe or unsafe condition so that there are four possible combinations of output. One of these—both showing safe—is taken as indicating safe, and the other three are taken as indicating unsafe. It is, however, possible to so arrange the system that both instruments or subsystems must show unsafe before the reactor is scrammed. This increases the reliability but reduces the safety and would be suitable for trip instruments only when the consequences of unsafe failure were not too severe.

The next stage is to use three instruments or subsystems, which makes possible the use of the principle of coincidence. This principle acknowledges the old saying that "two heads are better (more reliable) than one." If any one instrument can actuate the safety devices, the reactor can be scrammed unnecessarily because of the maloperation of one instrument. The reliability of the reactor system can be increased, however, by requiring that two or more of the instruments must agree that an unsafe condition exists before the safety devices are tripped. If any two signals (and this can occur in three different ways) that agree are accepted as correct, both safety and reliability are greatly improved. Figure 2 shows a system of this type.

Built-in Testing

An additional advantage of the application of the principle of coincidence to reactor control circuits is the possibility of using a built-in testing scheme. Since at least two out of three instruments must indicate an unsafe condition before the reactor is scrammed, such a condition can be simulated for each instrument in turn to check that it and its corresponding trip circuit are operating properly. If the testing capability is built into each trip circuit

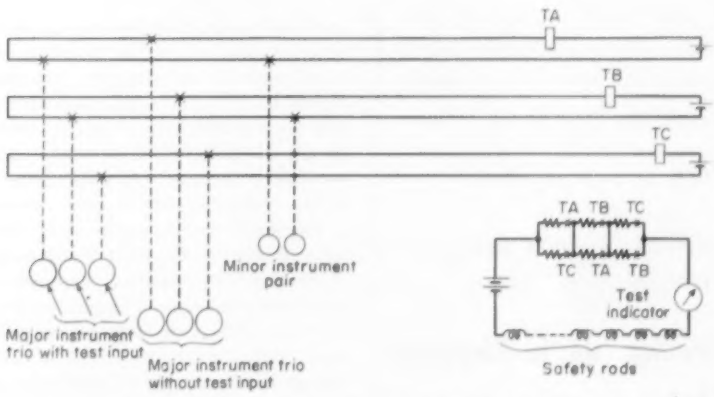
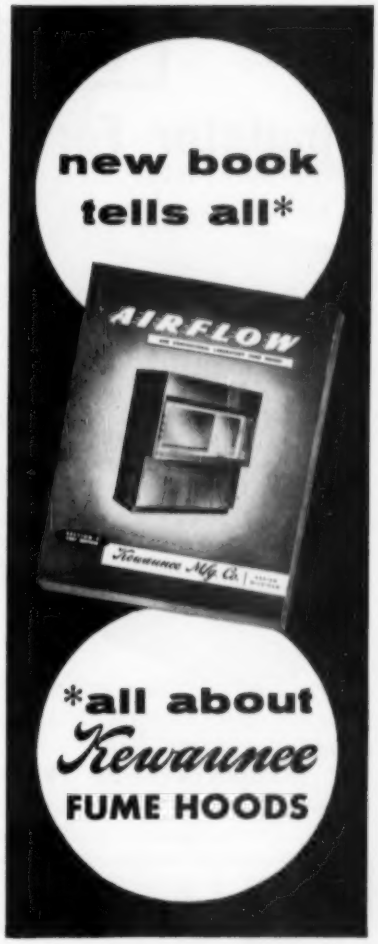


FIG. 2. Triple protective system is next stage in evolution. It employs both principle of redundancy and of coincidence—two of three instruments must show warning at same time to trip. Safety rods, however, remain in single circuit



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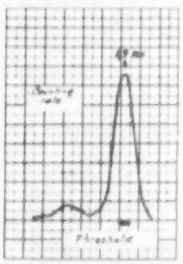

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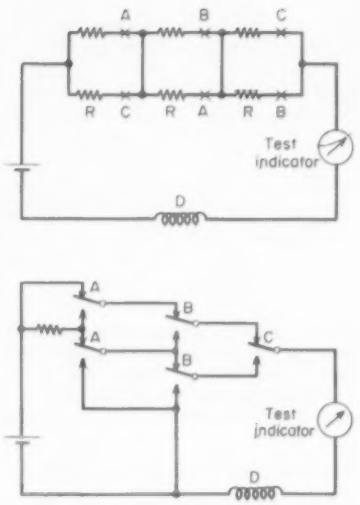

FIG. 3. Two examples of coincidence relay circuits designed to permit over-all testing of important functions

this important information can be acquired by the reactor operator at any time—even during reactor operation. Thus, both the safety and the reliability of the reactor system is increased.

If the coincidence-relay circuit is arranged as are the two cases shown in Fig. 3, a searching check on the whole system can be applied. In both circuits, the element D will be de-energized if any two of the relays A, B and C are released. In the upper circuit, a small resistance R is included in the circuit in series with each A, B or C relay contact, and an ammeter shows the current in the circuit. With A, B and C all closed, the total resistance in the circuit is 1.5 R. If one of the three relays opens, the resistance increases to 2.5 R.

The value of R is chosen so that this increase reduces the current by 10%, which would be insufficient to release the element D. If, then, the test condition is applied to an instrument, and the required 10% drop is observed in the final circuit, it is almost impossible for there to be an unsafe failure in the system up to the final circuit.

The lower circuit has the additional advantage of shorting out the element D when it is de-energized, thus reducing the chance that it can be re-energized by deliberate external connection. Also, only one resistance is required, and the normal resistance of the network is zero.

The tests do not involve any disconnection or disturbance of the equip-

ment and no expert knowledge is needed to carry them out. If the method of applying the test becomes faulty, the protection is in no way diminished. Since the correct result will not be obtained, the matter cannot go unnoticed.

In some applications it may be worthwhile to carry out these tests automatically at regular intervals, an alarm being sounded if the correct result is not received. Such a system would correspond to the "dynamic fail-safe" system* except that the protection would not be lost even during the test period of the cycle.

The weakest part of a system, such as that shown in Fig. 2, is the single circuit leading to the safety rods. There is, also, another weak point in that although three independent circuits are used, they are identical. It would involve little more effort to deliberately disable two of them, which would disable the whole system, than to disable one.

Figure 4 shows a scheme that eliminates this weakness. Each of the six safety devices has its own coincidence-relay group connected to the trip lines, and each has its own test ammeter as before. The principal protective function is covered by six instruments, three each of two different types to minimize the chance of disablement because of a common design or maintenance failure. Three of the instruments are connected to three main trip lines. The additional three are connected to an auxiliary trip line that is kept physically separate from the main one and controls only two of the safety rods and one dump valve. This is all that is needed, since the combined probability of the safety rods and main trip lines failing at the same time is several orders of magnitude less than other risks.

To take full advantage of this system, the safety devices used should be self-restoring, that is, if the danger signal is removed from the device, no further reduction in reactivity should take place, and it should be possible to restore the device to its normal operating position. This would enable a test to be applied to each device in turn by applying the danger signal momentarily. The application of the danger signal by the test switch could be checked on the same ammeter that was used to check the response to the single danger signal injected at the

* W. G. Rowell, A. B. Van Rennes, *Control Engineering* **3**, No. 3, 79 (1956).

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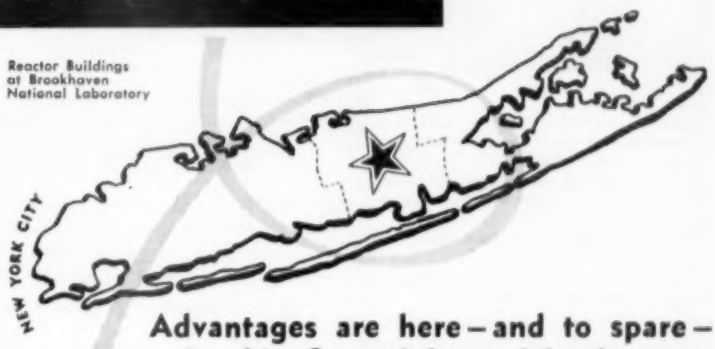

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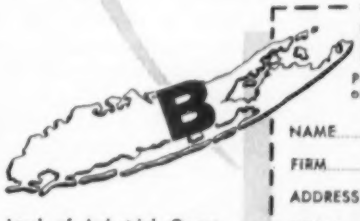

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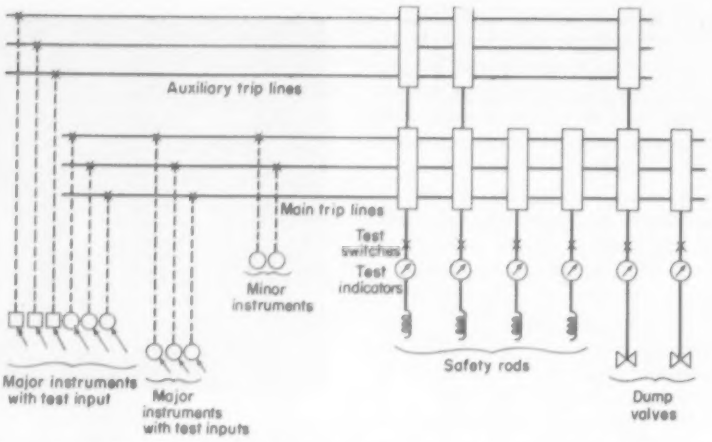

FIG. 4. This multiple-circuit protective system uses up to six instruments in double two-out-of-three coincidence to provide two independent circuits for safety devices

beginning of the protective channel. The logic of the overall test is then complete, and the probability of the system passing these tests and then failing to respond to genuine trouble is extremely remote.

A further possibility of increasing reliability also arises if the scram capacity is divided into reasonably small units such as often occurs when safety rods are used. It may then be possible to restart the reactor even after one safety rod has wrongly dropped into the core provided that this is done immediately—before the xenon poison has built up. This result can be achieved if the normal control system has enough reactivity in reserve to overcome the negative effect of one saftey rod.

If this possibility can be realized, a great increase in reliability will result, since those parts of the system which are not in a coincidence scheme are the ones that would normally be expected to cause most reactor shutdowns.

Significance

It is difficult to deal generally with the quantitative comparison between the simple single and double protective system. It can be seen, however, that if the system suffers n unsafe faults per unit time at random, and the average time to detect an unsafe fault is t, the single system is disabled for nt of the time and the double system for $(nt)^2$ of the time (if $nt \ll 1$). Now n may typically be 2/yr and t perhaps 10 hr, or 10^{-3} yr, so that $nt = 2 \times 10^{-3}$ and $(nt)^2 = 4 \times 10^{-6}$.

If n_1 is the frequency with which trouble arises that requires the protective system to function, the chance of accident is $n_1\ nt$ for the single system and $n_1\ (nt)^2$ for the double system. If n_1 is taken as 2/yr, the probability of accident is 4×10^{-3}/yr for the single system and 8×10^{-6}/yr for the double system. Thus, the chance of an accident is seen to be reduced by a factor of 500.

Some of the more detailed quantitative implications are shown in the table on p. 124. The figures given are typical and indicate the very great improvements in reliability that can be obtained. These calculations are, of course, much simpler than would apply in the case of practical networks in which the safety criteria for different instrument groups may vary widely. The improvements obtained depend entirely on the randomness of the failures. If there is any chance of correlation between failures on different instruments or subsystems, the advantage can be lost. It is necessary, therefore, to ensure that causes of such coupling are minimized. For instance, a reasonable degree of physical separation is desirable, and separate power distribution and, if possible, separate power sources, should be used for the different subsystems.

Application

The principles of redundancy and coincidence have already been applied to some extent in the NRX reactor. The safety of this reactor is provided by a bank of six safety rods and a set of six large valves that can drain out the heavy-water moderator. This equipment is controlled by several independent electrical circuits, so that the chance of simultaneous failure of all the safety devices is impressively

remote. Two groups of electronic safety instruments have been arranged in two-out-of-three coincidence to virtually eliminate false trips from these particular sources.

The new NRU reactor will use these same principles extensively. Almost every instrument group that is capable of tripping the reactor is arranged in a two-out-of-three– or two-out-of-four-coincidence connection, and this feature is carried through the entire electrical circuitry. Owing to the rather belated introduction of the new philosophy, however, it has not yet proved possible to arrange for the injection of test signals as a routine operation. This must await later reactors.

It is hoped that these two reactor-safety systems will provide working experience with which these new philosophies can be judged. The somewhat higher capital cost should prove to be more than compensated by the value of the improved safety and reliability that will result.

* * *

This article is based on a paper presented at the Canadian IRE Convention at Toronto in October, 1956.

Calculating Burnout Effects

By ARTHUR W. BANISTER* and RAYMOND L. MURRAY
Physics Department, North Carolina State College, Raleigh, North Carolina

We have developed a way to analyze the effect of fuel burnup and distributed poisons in a thermal reactor that we believe is simpler and easier to apply than conventional perturbation theory (*1*). Briefly, the technique involves an interative solution of the critical diffusion equation that describes the reactor after fuel burnup or fission product buildup has occurred. The terms introduced by perturbations are transposed to the right-hand side of the equation and by making well-justified approximations, the equation is rendered inhomogeneous. Complementary and particular solutions are found that give the new flux distribution and, by applying boundary conditions, the perturbed critical condition may then be obtained.

The most important physical changes occurring during the long-term operation of nuclear reactors are burnup of fuel and formation of fission products. In highly enriched reactors, the burnup of U^{235}, formation of plutonium and changes in moderator or structural materials also occur but can be shown to be negligible.

The differential equations describing the situation of interest are

$$\frac{dN_{235}}{dt} = -\phi\Sigma_{235} \qquad (1)$$

$$\frac{dN_p}{dt} = -\phi\Sigma_p + \phi\Sigma_{f235} \qquad (2)$$

where p refers to fission products. These yield standard exponential solutions, where ϕ may be treated either as a constant or variable with time, depending on the accuracy desired.

Critical Criteria

The effect of these variations in concentration on the chain reaction or flux distribution must be obtained from the diffusion equation, subject to certain conditions. The nature of these conditions merits some discussion. First, it is clear that a reactor will soon go subcritical as fuel burns out and fission products form, unless some compensating control is effected to keep the reactor critical. The criterion by which adjustments are made must be considered.

In a uniform low power reactor, where composition remains constant, it is evident that the critical reactor satisfies three conditions

$$\frac{\partial\phi}{\partial t} = 0$$

constant local flux density (3)

$$\frac{\partial}{\partial t}\int \phi\, dV = 0$$

constant total neutron population (4)

$$\frac{\partial}{\partial t}\int \phi\Sigma_f\, dV = 0$$

constant total reactor power (5)

In a high-flux reactor, however, where composition is changing appreci-

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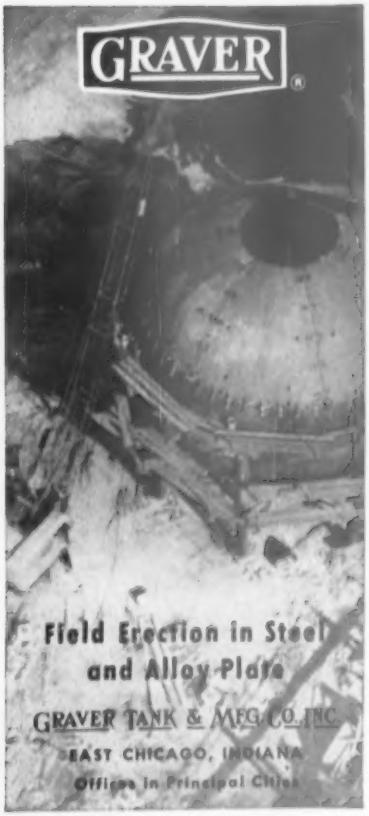


ably with time, each of these criteria implies something slightly different. Condition (3) is clearly not compatible with the continuous change in composition indicated by (1) and (2), since group constants in the diffusion equation are continually changing and assuming new spatial dependence. Thus, $\partial\phi/\partial t$ cannot possibly be zero at all points in the reactor. After the reactor has been operated for a certain interval in accordance with condition (5), the fuel and poison distribution will be non-uniform, and the flux will be distorted from its initial shape. Since the changes in composition, flux shape and amplitude are all slowly varying with time the neutron balance condition based on equality of supply and leakage plus absorption still is satisfied approximately. Thus condition (3) may be applied instantaneously.

To demonstrate the method the effects of fuel burnout on flux distributions are derived for a bare-enriched homogeneous slab reactor. The primary processes studied are burnout of U^{235} and formation of low cross-section fission products. Both transient and steady-state xenon poisoning are excluded. Also, it is assumed that control absorbers are uniformly removed to keep the reactor critical.

First, it is necessary to write the diffusion equations before burnout.

$$D_1\nabla^2\phi_1 - \Sigma_1\phi_1 + \frac{k}{p}\Sigma_2\phi_2 = 0 \quad (6)$$

$$D_2\nabla^2\phi_2 - \Sigma_2\phi_2 + p\Sigma_1\phi_1 = 0 \quad (7)$$

Uncoupling by the operator method, one obtains

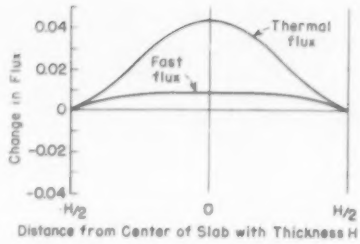
$$(\nabla^2 + \mu^2)(\nabla^2 - \nu^2)\phi_1 = 0$$
$$(\nabla^2 + \mu^2)(\nabla^2 - \nu^2)\phi_2 = 0$$

where μ^2 and ν^2 are the usual roots of the critical buckling equation. The solutions to these equations for the bare slab geometry are

$$\phi_1 = A\cos\mu x$$
$$\phi_2 = S_1 A\cos\mu x$$

Now assume that the reactor has been operating for a period of time during which fuel consumption has occurred, and absorber poison has been removed. The principal changes will be in k and Σ_2, since all other group constants depend on scattering cross sections, which remain essentially constant. Assume further that changes are halted for the instant under consideration, so that criterion (3) may be applied. The new diffusion equations are

$$D_1\nabla^2\phi_1 - \phi_1\Sigma_1 + \frac{k}{p}\phi_2\Sigma_2 - \eta\phi_2\Delta\Sigma_u(x) = 0 \quad (8)$$



FLUX CHANGE resulting from fuel burnup in bare slab reactor. Changes are relative to initial flux at reactor center. Burnup at center was 5%

$$D_2\nabla^2\phi_2 - \phi_2[\Sigma_2 - \Delta\Sigma_u(x) - \Delta\Sigma_p] + p\Sigma_1\phi_1 = 0 \quad (9)$$

where $\Delta\Sigma_u$ is the uranium burned out and $\Delta\Sigma_p$ is the absorber poison removed.

The elements introduced by burnout are now transposed to the right hand side of the equation. For short times of operation these terms may be approximated by assuming the thermal flux equal to its initial value. The change in Σ_u is proportional to this flux. Thus one may write for the slab geometry

$$D_1\nabla^2\phi_1 - \phi_1\Sigma_1 + \frac{k}{p}\phi_2\Sigma_2 = \eta S_1 A\Delta\Sigma_{uc}\cos^2\mu x$$
$$D_2\nabla^2\phi_2 - \phi_2\Sigma_2 + p\,\phi_1\Sigma_1 = -S_1 A(\Delta\Sigma_{uc}\cos^2\mu x + \Delta\Sigma_p\cos\mu x)$$

where $\Delta\Sigma_{uc}$ is the change in Σ_u at the center of the reactor. Uncoupling again by the operator method, one obtains

$$(\nabla^2 + \mu^2)(\nabla^2 - \nu^2)\phi_1 = \Delta\Sigma_{uc}H_1 A + \Delta\Sigma_p L_1 A\cos\mu x + \Delta\Sigma_{uc}M_1 A\cos 2\mu x$$

and

$$(\nabla^2 + \mu^2)(\nabla^2 - \nu^2)\phi_2 = \Delta\Sigma_{uc}H_2 S_1 A + \Delta\Sigma_p L_2 S_1 A\cos\mu x + \Delta\Sigma_{uc}M_2 S_1 A\cos 2\mu x$$

where

$$H_1 = \frac{k - \eta p}{2D_2 k}\left(\mu^2 + \frac{\Sigma_1}{D_1}\right)$$

$$L_1 = \frac{1}{D_2}\left(\mu^2 + \frac{\Sigma_1}{D_1}\right)$$

$$M_1 = \frac{1}{2D_2 k}\left(\mu^2 + \frac{\Sigma_1}{D_1}\right)\left(k - \eta p - \frac{4\eta p\mu^2 D_2}{\Sigma_2}\right)$$

and

$$H_2 = \frac{\Sigma_1(1 - \eta p)}{2D_1 D_2}$$

$$L_2 = \frac{1}{D_2}\left(\mu^2 + \frac{\Sigma_1}{D_1}\right)$$

$$M_1 = \frac{1}{2D_2}\left[\frac{\Sigma_1}{D_1}(1-\eta p) + 4\mu^2\right]$$

These equations may now be solved by obtaining complementary and particular solutions. The complementary solution is obviously the original flux while the particular solution represents the perturbations caused by burnout. Before obtaining the particular solution, it is convenient to approximate the operator $(\nabla^2+\mu^2)(\nabla^2-\nu^2)$ by $-\nu^2(\nabla^2+\mu^2)$. This simplifies the operations required, and is also justifiable mathematically, since ∇^2 operating on a μ-type term will usually give a quantity much less than ν^2.

With this simplification, the solutions to the diffusion equations become

$$\phi_1 = A\cos\mu x + \frac{A(1+\mu^2\tau)}{2\Sigma_2(k-1)}\left[\Delta\Sigma_{ac}\left(\frac{1}{f}-1\right)(1-\tfrac{1}{3}\cos 2\mu x) - \frac{4\mu^2L^2}{3f}\Delta\Sigma_{ac}\cos 2\mu x - \Delta\Sigma_p\mu x\sin\mu x\right] \quad (10)$$

$$\phi_2 = S_1A\cos\mu x + \frac{S_1A}{2\Sigma_2(k-1)}\left[\Delta\Sigma_{ac}(\eta p-1)(1-\tfrac{1}{3}\cos 2\mu x) + \frac{4\mu^2\tau}{3}\Delta\Sigma_{ac}\cos 2\mu x - \Delta\Sigma_p(1+\mu^2\tau)\mu x\sin\mu x\right] \quad (11)$$

Utilizing the fact that both fluxes must still go to zero at the reactor boundary, the critical conditions are obtained

$$\Delta\Sigma_p = \frac{8\Delta\Sigma_{ac}}{3\pi}\left(\frac{1+\mu^2L^2}{f}-1\right) \quad (12)$$

$$\Delta\Sigma_p = \frac{8\Delta\Sigma_{ac}}{3\pi}\left(\frac{\eta p}{1+\mu^2\tau}-1\right) \quad (13)$$

These can be shown to be identical. Thus, we have obtained the new flux distributions and the required removal of poison in rather simple form.

As can be seen from the figure the thermal flux peaks at the center, while the fast flux flattens. Both fluxes also rise slightly; this is due to the changes in cross section required to keep the reactor critical. The following relations may also be derived from (12) and (13):

$$\frac{\Delta\phi_{1c}}{\phi_{1c}} = \frac{\Delta\Sigma_{ac}}{\Sigma_a}\frac{(1+\mu^2\tau)(1-f-2\mu^2L^2)}{3(k-1)}$$

$$\frac{\Delta\phi_{2c}}{\phi_{2c}} = \frac{\Delta\Sigma_{ac}}{\Sigma_a}\frac{f(\eta p-1+2\mu^2\tau)}{3(k-1)}$$

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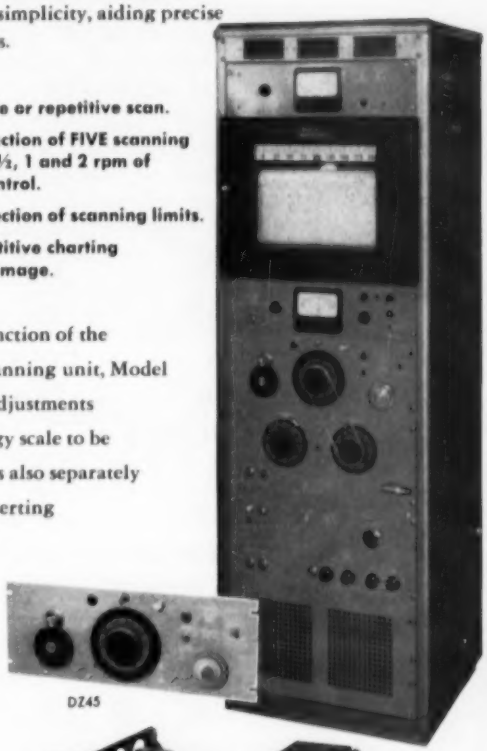

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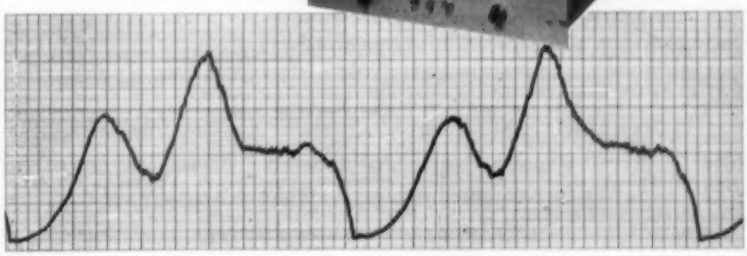

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These relate the change in central flux to the change in central fuel cross section. Using group constants typical of the Bulk Shielding Facility at Oak Ridge, it can be shown that the thermal flux changes some five times faster than the fast flux. The peaking of the thermal flux is also significant since it tends to offset the decrease in fission cross section. Thus the power tends to remain approximately constant.

So far fission product effects have been neglected. However, these may be treated in exactly the same fashion, either simultaneously or separately. Qualitatively, it can be seen that this effect will tend to flatten both fluxes, and increase the poison removal required.

Comparison with Perturbation Theory

To provide a cross check a comparison has been made of the critical conditions obtained by the preceding analysis and standard perturbation theory. This is illustrated briefly for the two group bare reactor.

The critical condition from first order perturbation theory (*1*) can be shown to be:

$$\int \phi^+ \delta H \phi \, dV = 0$$

where ϕ^+ is the adjoint flux and δH is the perturbation operator. For a two group reactor, this is generalized to matrix form as follows:

$$\int (\phi_1^+, \phi_2^+) \begin{pmatrix} \delta H_{11} \delta H_{12} \\ \delta H_{21} \delta H_{22} \end{pmatrix} \begin{pmatrix} \phi_1 \\ \phi_2 \end{pmatrix} dV = 0$$

From (8) and (9), it is evident that $\delta H_{11} = 0$, $\delta H_{12} = -\eta \Delta\Sigma_u(x)$, $\delta H_{21} = 0$, and $\delta H_{22} = \Delta\Sigma_u(x) + \Delta\Sigma_p$. Also, it can be shown that:

$$\phi_1^+ = A^+ \cos \mu x$$
$$\phi_2^+ = S_1^+ A^+ \cos \mu x$$

Substituting these expressions, expanding, and dividing out common factors, one obtains the critical condition for a bare slab:

$$-\eta \Delta\Sigma_{uc} \int \cos^3 \mu x \, dx + S_1^+ \Delta\Sigma_{uc} \int \cos^3 \mu x \, dx + S_1^+ \Delta\Sigma_p \int \cos^2 \mu x \, dx = 0$$

Using the fact that $S_1^+ = (1 + \mu^2\tau)/p$, this simplifies to:

$$\Delta\Sigma_p = \frac{8\Delta\Sigma_{uc}}{3\pi} \left(\frac{\eta p}{1 + \mu^2 \tau} - 1 \right)$$

which is identical with (12) and (13). Thus, for this simple reactor, the cross check is exact.

To date, this type of analysis has

been conducted for the bare slab using modified one group, age-diffusion, and two-group theory, for the reflected slab using modified one-group and two-group theory, and for the bare sphere using modified one-group theory. Preliminary analysis of the spherical and cylindrical geometries indicates that the necessary operator theorems may be derived, using known differential equation techniques. The variation of parameters method has been found to be particularly valuable.

In the case of the reflected reactor, the necessary relation between $\Delta\Sigma_{uc}$ and $\Delta\Sigma_p$ for criticality by the differential equation method is found to differ somewhat from the result obtained by perturbation theory. Both expressions, however, are very similar, and produce similar numerical values. For example, using group constants typical of the BSF, the following results were obtained:

differential equation method

$$\Delta\Sigma_p = 0.465\Delta\Sigma_{uc}$$

perturbation theory

$$\Delta\Sigma_p = 0.423\Delta\Sigma_{uc}$$

The difference may be ascribed to the fact that the latter does not take into account changes in the reflector flux distribution.

Conclusions

In summary we suggest that the differential equation analysis has the following advantages:

(1) Mathematical simplicity, with no knowledge of perturbation theory required.
(2) For the same degree of accuracy of iteration, the perturbed flux distributions are obtained in closed form rather than the infinite series characteristic of perturbation theory.
(3) The effect of each physical perturbation on the flux distribution can be identified.
(4) The critical condition for the two-group reflected reactor takes account of the changing fluxes and currents at the core-reflector interface.
(5) A wide flexibility, with the possibility of application to such other problems as xenon buildup, non-uniform coolant distributions and plutonium production.

BIBLIOGRAPHY

1. S. Glasstone and M. C. Edlund, "The Elements of Nuclear Reactor Theory" (D. Van Nostrand Company, Inc., New York, 1954)

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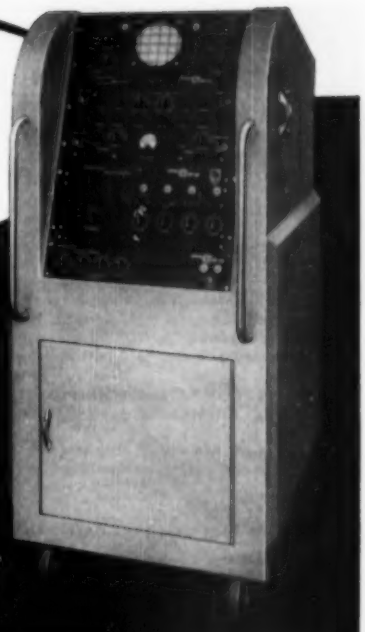

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Small Reactor Is Analytical Tool

The effect of a sample on the reactivity of a small reactor can provide a sensitive; nondestructive analysis of the constituents of the sample.* The method is an extension of the usual control-rod calibration in which the effect of a known substance on reactivity is studied. By observing the effects at different points in the reactor it is possible to separate different neutronic properties. Thus one can measure not only nuclear fuels and poisons but also such quantities as the carbon-hydrogen ratio of petrochemicals and plastics.

How it works. The effect of a sample on reactivity is a function of its neutronic properties: scattering, absorption, moderation, production of neutrons, etc. It depends also on the position of the sample in the reactor, and this dependence on position is different for different neutronic properties. Thus a scatterer is most effective when it is at the edge of the core where it will reflect back into the core some of the neutrons that would otherwise leak out. In contrast a thermal absorber is most important where the thermal flux is high.

In general it is possible to define a set of "importance functions" to represent the relative effect that a "pure" sample has on the reactivity when it is in different locations in the core. These functions are products of two factors, one to represent the frequency of an event, and the other to represent the magnitude of the influence of the event on reactivity. As an example, thermal absorption has an importance function nearly proportional to the square of thermal flux. One factor comes from the fact that the number of captures is proportional to flux. The other results from the direct proportionality between thermal flux and thermal-neutron usefulness.

Analysis. It is possible to describe the influence of the sample on reactivity with the following equation $\Delta\rho = \sum_i C_i I_i(x) \Delta\Sigma_i$ where Σ_i is the total area presented by the sample for the ith process (macroscopic cross section × volume), I_i is the importance function normalized to unity at the point of maximum importance, and C_i is a reactivity coefficient.

As indicated above, the importance functions can be computed. Using two neutron groups, fast and thermal, they can be represented with fair accuracy in terms of fluxes and gradients of fluxes. They can also be measured by making traverses of the reactor with appropriate test samples and recording the effect on reactivity. Their representations and the methods of measuring them are as follows:

Effect	*Importance function*	*Measured by traverse with*
Thermal absorption	ϕ_t^2	Thermal absorber
Fast absorption	ϕ_f^2	Fast absorber (e.g., U^{238})
Scattering	$A(\nabla\phi_t)^2 + B(\nabla\phi_t)^2$	Heavy scatterer
Moderation	$\phi_t\phi_t$	Light moderator
Neutron production (fission)	$\phi_t\phi_t$	U^{235}

When two effects occur together it may be necessary to find one of them by subtracting the other. Once the C's and I's are determined, a set of simultaneous equations can be set up for any sample by measuring its influence on reactivity at several positions. From these one can solve for the important Σ's.

Reactor characteristics. For analyses of this type it is recommended that a reactor have the following properties:

1. Power level >10 milliwatts to reduce statistical fluctuations.
2. Power level <1 watt so that thermal effects on reactivity are small.
3. Small critical mass to make the C's large.

A homogeneous aqueous reactor fits these criteria best, but for some measurements it is also important to have a high ratio of thermal to fast flux.

It is believed that a small reactor like the AGN-201 is capable of recognizing 11 μg of boron, 31 μg of cadmium, and 1.14 mg of hafnium with 10-min observations. An observation of this length produces optimum sensitivity.

Carbon-hydrogen ratio is particularly susceptible to measurement by this means because of the large moderating power of hydrogen as compared with that of carbon. With observations of 3 min, the ratio of the number of hydro-

* R. Boyle *et al.* The AGN 201 nuclear reactor as an analytical instrument: process control. (Aerojet-General Nucleonics, Report No. 12, January, 1957)

gen atoms to the number of carbon atoms in a petrochemical can be determined to within about 0.03 in the H/C ratio.

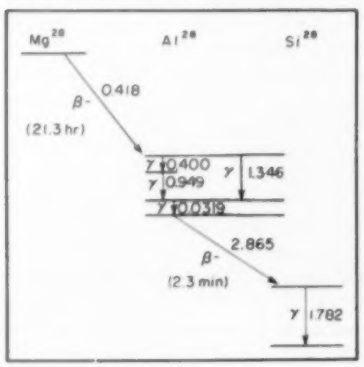
New Process Makes High-Activity Mg^{28} Magnesium is an important industrial metal and an important element in plant and animal metabolism. Chemical analysis for small amounts of it is difficult, however, and tracer experiments have been hampered by the difficulty of producing the only magnesium isotope with a half-life longer than the 9.5 min of Mg^{27}. At Brookhaven L. G. Stang, Jr., W. D. Tucker, A. J. Weiss, and H. O. Banks, Jr., have now developed a procedure that produces 21.3-hr Mg^{28} with a specific activity $>1{,}000$ μc/gm, essentially free of impurities.

The process consists of making a thin foil of lithium-magnesium alloy and irradiating it with neutrons. Tritons are produced in the $Li^6(n,t)He^4$ reaction, and these in turn make Mg^{28} through the reaction $Mg^{26}(t,p)Mg^{28}$. Lithium enriched to contain $\sim$96% Li^6 is combined with natural magnesium in an alloy containing 70% lithium by weight. Enrichment of the Mg^{26} content would be useful but forbiddingly expensive.

To avoid depressing the neutron flux the foils are made as thin as is practical —0.01 in. Irradiation is for 4 days in a helium atmosphere and a flux of 3.5×10^{12} n/cm^2/sec. After irradiation chemical processing takes $\sim$2 hr and produces $\sim$95% yield.

The decay scheme of Mg^{28}, which exists in equilibrium with its daughter 2.3-min Al^{28}, is shown in the drawing. Magnesium-28 is shipped in a disposable unshielded container. Production runs are being made weekly at Brookhaven, and the cost is $39.80 per 50-$\mu$c unit.

DECAY SCHEME of Mg^{28}



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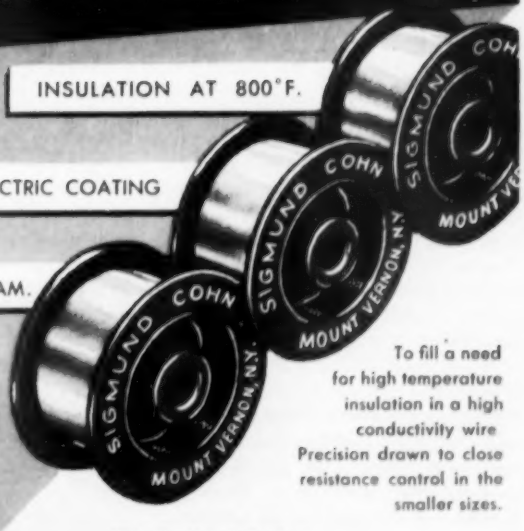

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C^{14} Tracer Measures Fuel Distribution

By DOUGLAS E. COOPER
Ethyl Corporation Research Laboratories
Ferndale, Michigan

The performance of modern automotive engines is noticeably limited by the failure of the intake system to supply equal amounts and qualities of fuel to all the cylinders. This maldistribution can be minimized by fuel injection, but injection requires expensive equipment. Maldistribution involves mainly the liquid portion of the charge. Normally the inducted fuel is only partially evaporated; the remaining liquid is carried in the form of entrained droplets and swept along the inside walls of the manifold as a liquid sheet. Since this liquid consists only of the less volatile constituents of the gasoline, the cylinders getting more of this material are burning fuel not only in greater quantity, but also of different over-all quality.

C^{14} Method

In this situation it is desirable that such additives as tetraethyl lead (TEL), scavengers, and phosphorus compounds be divided among the cylinders in such a way as to provide maximum effectiveness. A C^{14}-tracer technique has been developed for determining how any constituent is distributed, whether it is a gasoline hydrocarbon or an additive. For example, TEL tagged with C^{14} at a level of 0.05 mc/cm³ can be readily traced after addition to fuel in a typical concentration of 3 cm³/gal.

When such a fuel is burned in an engine, the various concentrations of TEL as received by individual cylinders can be determined by comparing the specific activities of the carbon dioxide in their combustion gases. These activities are measured in an ionization chamber of the continuous-current type.

Sample Collection

The combustion gases in each cylinder are sampled by replacing the conventional spark plugs by specially developed sampling valves. The nose of this device, shown in Fig. 1, contains a spark-plug gap and also a tiny poppet valve, actuated by a solenoid timed from the camshaft. The poppet opens for about 1 msec in each cycle, just after combustion is complete and before the exhaust valve opens.

The original intention was simply to dry the combustion-gas samples and introduce them into the chamber. However, for reasons discussed below it proved more satisfactory, and actually simpler, to retain only the carbon dioxide of the gas streams. For this purpose the gases are bubbled through barium hydroxide solutions. From the resulting barium carbonate, carbon dioxide is later regenerated and introduced into an ionization chamber. A vibrating-reed electrometer is used to assay the radioactivity.

In measuring the ionization current, complications appear if the whole combustion-gas mixture is used. In the first place, carbon monoxide, carbon dioxide, and nitrogen yield slightly different numbers of ion pairs per beta particle passing through them. Since these gases are present in varying proportions in the samples because of differences in the richness of the fuel-air mixtures, samples containing identical amounts of C^{14} can produce different currents. (This variable could be controlled by a calibration study, comparing the currents produced by a fixed quantity of C^{14} in a series of combustion-gas compositions representing the whole working range of fuel-air mixtures.)

A more serious problem is the effect of trace constituents. Many of these, such as nitric oxide, formaldehyde, and benzene, have very low ionization potentials as compared with the major gases. As shown by Jesse and others,

FIG. 1. Gas-sampling valve has spark-plug gap and solenoid-operated poppet

very small quantities of such substances can greatly increase the number of ion pairs produced in a less ionizable major constituent, in this case carbon monoxide, carbon dioxide and nitrogen.* This is because the principal gas always contains, along with the completely ionized molecules, large numbers of excited molecules having energies somewhat beneath their ionization levels. These excitation energies are more than adequate for ionizing such molecules as formaldehyde. Accordingly, by the simple process of energy transfer during intermolecular collisions, a small admixture of such a gas can yield a remarkably large number of additional ions. This effect is found to be very sensitive to concentration in just the particular low range in which these gases occur. Furthermore, since most of the offending gases happen to be quite reactive and labile, concentrations are unpredictable, and the whole situation is difficult to control with any confidence. This problem, as well as that arising from differences in the proportions of major gases, is avoided completely by discarding all the gases but carbon dioxide.

The sampling system installed on a late-model V-8 engine is shown in Fig. 2. Each cylinder is equipped with a sampling valve, which is connected to two spiral-path gas-washing bottles fed in parallel, providing duplicate samples.

* W. P. Jesse *et al*, *Phys. Rev.* **88**, 417 (1952); T. E. Bortner *et al*, *Phys. Rev.* **96**, 643 (1954).

Another pair of absorbers located underneath are connected to a bent probe located in the exhaust pipe in order to sample the over-all exhaust. The exhaust hood at the rear was installed to reduce the hazard from leaks of radioactive materials. The dynamometer itself is located behind the hood. When in use, the absorbers are filled with a 4% barium-hydroxide solution containing a *p*H indicator such as phenolphthalein. The engine is operated until all the samples are neutralized, with a system of stopcocks to divert the flows away from those that happen to finish first. Usually all the samples reach neutrality in 5–10 min. Later the precipitated carbonates are filtered, washed with water and alcohol, and then dried. They average 3–4 gm.

Ion-Chamber Filling

The carbon dioxide can be regenerated from these samples at the convenience of the operator with the train shown in Fig. 3. At the extreme left is the generator into which the dry carbonate is introduced. The entire path from generator to chamber (extreme right) is then evacuated, and the mercury in an automatic valve adjacent to the generator is allowed to rise and seal off the rest of the system. Concentrated sulfuric acid is commonly used to release carbon dioxide from barium carbonate, but the mercury valve makes it possible to use water and the somewhat volatile, but more desirable, perchloric acid while the vacuum is maintained in the rest of the system. Water is added through the funnel, and a magnetic stirrer produces a slurry. Slow addition of dilute perchloric acid then generates the gas. When the pressure reaches a level near atmospheric, the mercury in the valve is depressed enough to permit a small volume of gas to escape before pressure reduction seals it again. This process

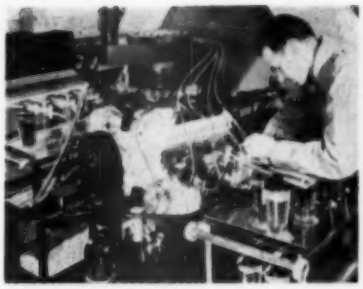

FIG. 2. In typical use each sampling valve leads to two spiral-path washing bottles

is repeated automatically as long as gas is evolved. If sulfuric acid were used, the coating of the particles by insoluble sulfate would rapidly slow down the reaction, but formation of the soluble perchlorate allows the reaction to proceed smoothly to completion.

In the first stage the gas is allowed to flow only as far as the U-tube near the middle of the train. During its passage the carbon dioxide is first passed through a Millipore filter in order to remove suspended particles, and then dried by a trap cooled with dry ice. It is then condensed to a solid in the U-tube chilled by liquid nitrogen.

This tube is then sealed off from the forepart of the system, and the air (from the reagents) is exhausted from the remainder. The liquid nitrogen is then removed, and the carbon dioxide sublimes into the ion chamber. The exact amount of the filling is recorded by a manometer. Additional ordinary carbon dioxide is added from a tank to bring the chamber pressure to some arbitrary level (typically 74 cm of Hg). This provides an identical ionization medium for all samples. The whole operation takes ~10–15 min.

Measuring Activity

The chamber is then detached and mounted on the electrometer. Figure

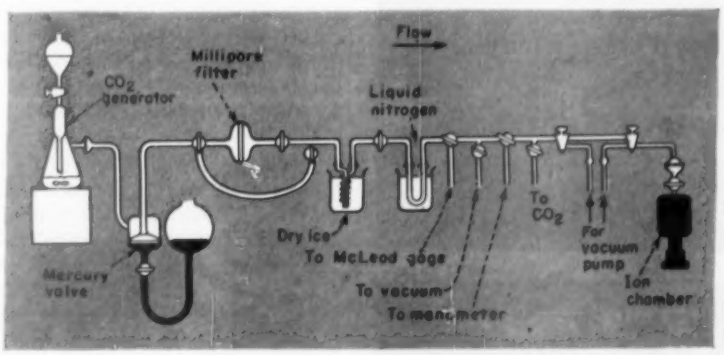

FIG. 3. This apparatus regenerates labeled CO_2 from $BaCO_3$ precipitate and passes it into ion chamber

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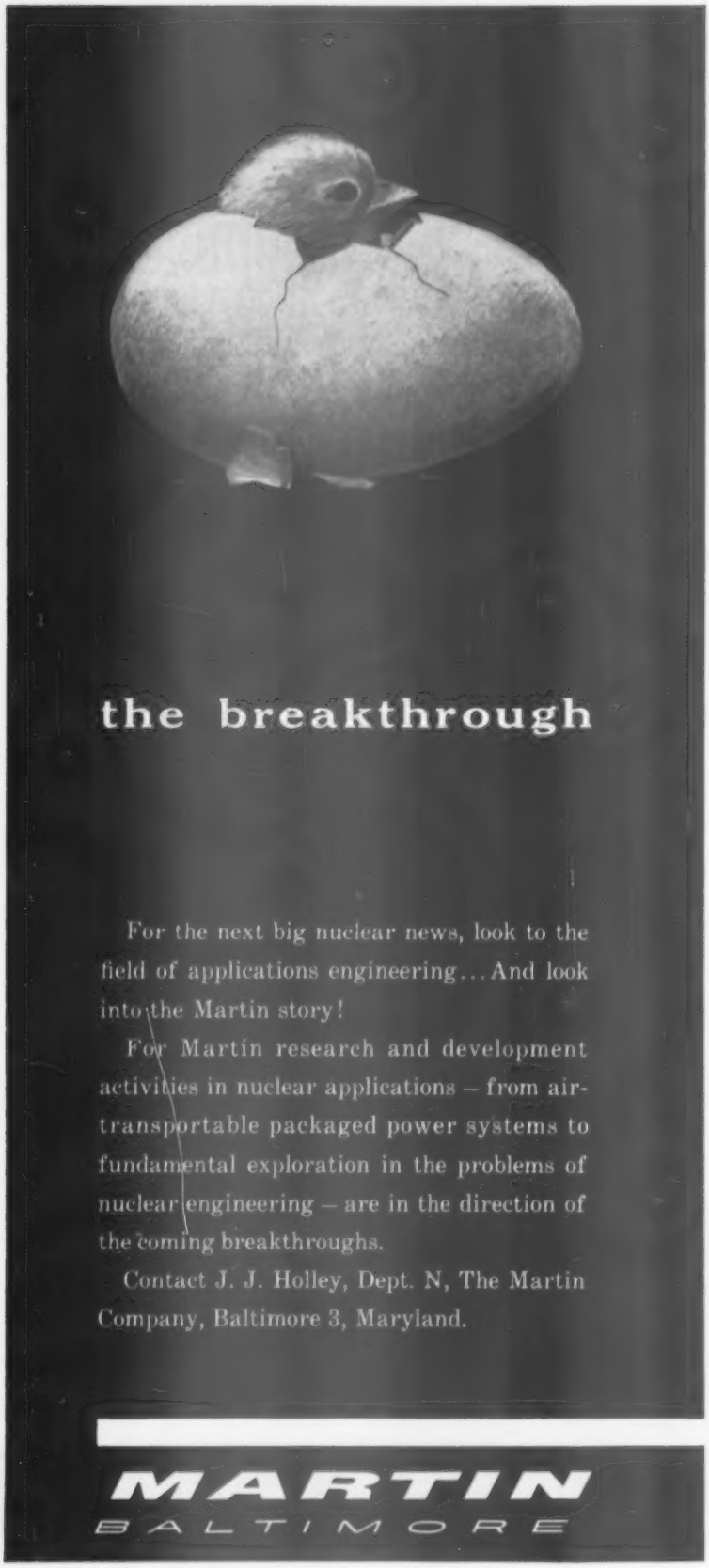
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FIG. 4. For counting, ion chamber is attached to vibrating-reed electrometer and slid into iron shield. Box on top is line-voltage regulator

4 shows the mounted chamber partially inserted in the iron shield. The shield reduces the background by 60%. It is made sectionally of 4 in. of iron, and is lined with $\frac{3}{8}$ in. of aluminum. Since the shield weighs nearly 300 lb, it was made in the form of a tunnel, and a sliding aluminum cradle provided to support the chamber and electrometer in a horizontal position.

An average of perhaps 1,000 ion pairs results from each C^{14} beta particle. The ion current through the carbon dioxide in the chamber is $\sim 10^{-14}$ amp. To obtain a perceptible effect from this, a charge is allowed to accumulate on a capacitance of about 10^{-11} farad, located in the electrometer. Part of this capacitance is a quartz reed vibrating at 475 cps. The reed produces a modulated voltage signal from the charge, permitting the use of a-c amplification. The amplified a-c voltage is rectified and filtered in auxiliary equipment, and then it actuates a strip-chart voltage recorder. The accumulation of voltage with time is ultimately used as a measure of the chamber current, and thus of the specific activity of the sample.

Figure 5 shows a typical record from a fairly weak source of activity. The period shown is just before the voltage-accumulation rate, or the current, reaches a steady minimum, which requires ~15 min.

The sharp discontinuities are caused by random alphas from the walls of the chamber. (Even the most carefully selected brass contains traces of certain alpha-emitting heavy metals.) As compared with perhaps a thousand ion pairs produced by a C^{14} beta, an alpha particle may produce a hundred thousand and a corresponding sharp increase in the accumulated charge.

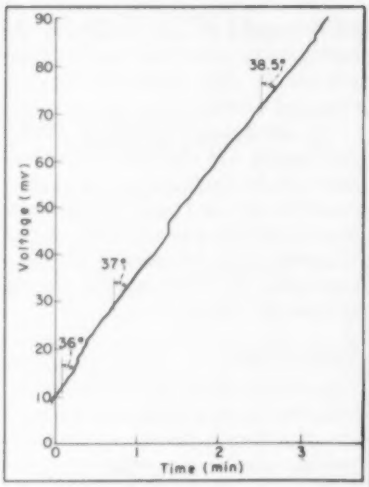

FIG. 5. Counting record from weak source shows discontinuities at points where alpha particles are emitted from chamber walls

After the slope becomes constant, a measure of the current can be obtained by observing the time required to accumulate any given voltage, or by simply measuring the slope of any straight section of this line and computing its reciprocal. Comparison of the current with that produced by a calibrated reference source gives the specific activity of the sample.

Isotopic Enrichment

It could not simply be presumed that the ratio of C^{14} to C^{12} atoms in the carbon dioxide would be exactly representative of the whole gas containing both carbon-monoxide and carbon-dioxide activities. To check the procedure a number of carefully controlled tests were run. The main objective was to establish whether the specific activity of the carbon dioxide as recovered from any given cylinder was exactly proportional to the concentration of tagged TEL received in the fuel. It also was necessary to show that this relationship was not affected by the variations in the richness of the mixture of fuel and air that are inevitable in actual distribution studies.

For this purpose one cylinder of the test engine was blocked off from its passage along the intake manifold and fed special fuels by way of a small special carburetor. The other cylinders were maintained on the stock carburetor using regular gasoline. Fuels were made up containing several concentrations of tagged TEL, and the test cylinder was fed with each of these at several fuel-air ratios. This cylinder alone was sampled with a pair of ab-

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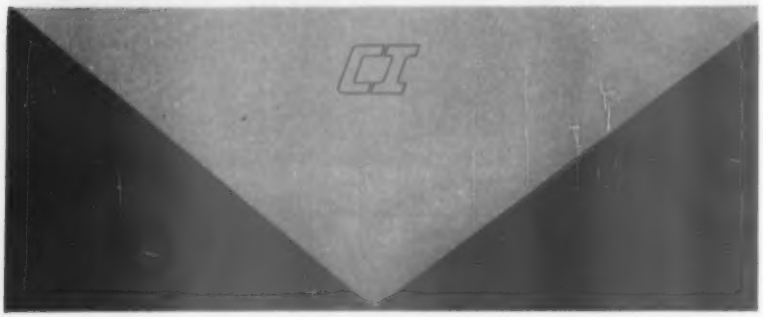

sorbers to obtain duplicate samples at each operating condition. The barium carbonate samples were processed and assayed by the standard procedure discussed previously.

No significant difference in C^{13}-C^{14} distribution was observed for carbon-monoxide fractions in the exhaust gases from 0 to 63% and for label concentrations from 2.0 to 3.8 cm³/gal. A plot of measured activity against TEL concentration gave a straight line passing through the origin.

Typical Test

In an actual test run, where the distribution to all cylinders is followed, the data always provide a number of internal checks on reliability. The table below shows the results of a TEL-distribution study. The eight cylinders are listed in two groups, as fed from the two barrels of the carburetor. The carbon dioxide from the combined exhaust shows an activity of 305 dps/gm. This represents the activity produced from the fuel being fed, which contains 3 cm³ of TEL per gallon.

Accordingly the activities found in individual cylinder samples of carbon dioxide can be translated proportionately into concentrations of TEL in the fuel charges of their cylinders. It will be noted that the TEL concentrations ranged from 2.55 to 3.72 cm³/gal in this test. Since it was carried out at nearly wide-open throttle but at low speed (800 rpm) this cannot be considered typical operation. Modern cars equipped with automatic transmissions can experience this combination of conditions only as a transient phase before attaining much higher speeds, at which distribution is considerably improved. However, the exaggerated maldistribution produced at the steady low speed can be useful in such tests by making the phenomenon easier to study.

TEL Concentrations

Cylinder	*Activity (dps/gm of CO_2)*	*TEL (ml/gal)*
1	381	3.72
4	252	2.46
6	261	2.55
7	301	2.95
2	348	3.40
3	305	2.98
5	274	2.68
8	288	2.82
Exhaust	305	3.00

TECHNICAL ADVANCES

High-Energy Physics—New Themes, New Hope Anyone who has followed the progress of high-energy nuclear physics over the past several years is aware that its principal problems revolve around an understanding of the "elementary" particles, more than a score of which are now known to exist. There is as yet no reason to believe that the list is complete. Nor do we know in very many cases whether or not we are permitted to remove quotation marks from the word "elementary" used to describe them. The situation is still closely analogous to the beginnings of chemistry, where it was not yet clear that only the hydrogen atom is an element in a comparatively strict sense, while all heavier atomic species are more properly regarded as combinations of hydrogen atoms and neutrons.

When it is finally reached, the achievement of an understanding of the subnuclear particles will have been made in the face of what seem to be far greater obstacles than any that stood in the way of an analysis of atoms. Many of the particles, including the more mysterious of them, are highly unstable; their mean lifetimes are often less than a millimicrosecond. Only very brief and rather rare glimpses of them can therefore be seen, and quantitative studies of their properties demand that they be produced at great expense in huge accelerators and examined in detectors of extraordinary ingenuity. There are additional difficulties of a theoretical nature, for the present viewpoint is that complex particles are not composed of other particles but of "fields," and even an elementary field is infinitely more complex than an elementary particle.

The atmosphere of the seventh Rochester conference on high-energy physics was in many ways considerably more optimistic than that of any of the earlier meetings. Part of this optimism arose from an appreciation of the great speed with which new experimental data are accumulating in laboratories throughout the world, despite the technical difficulties mentioned above. Another set of developments, appearing to reveal new vistas in theoretical physics, concerns the very recent observations indicating breakdown of invariance under space inversion (or interchange of "left" and "right") and charge conjugation in the weak interactions—specifically, in the decay of muons, pions, and nuclei.

Parity nonconservation. It seems at first paradoxical that the breakdown of long-cherished laws of physics can give rise to any sort of optimism, but in this connection we need only remind ourselves that the experimental overthrow of the concepts of absolute space and time led ultimately to the special theory of relativity. There is a note of irony, nonetheless, within the details of the recent developments. The ambiguous symmetry properties of K-mesons, which led Lee and Yang to their speculations on parity nonconservation, are still unexplained within the framework of theoretical ideas that account for nonconservation in the parity experiments of Wu and Lederman. Current ideas involve the so-called two-component theory of the neutrino, according to which neutrinos can be regarded as "right-handed" and antineutrinos as "left-handed." Since neutrinos are involved at each stage of the muon, pion, and beta decays so far examined, the theory can deal with these processes. But emission of neutrinos does not take place in the decay of mesons of type tau and theta, where the problem arose.

Although the results on parity nonconservation and noninvariance under charge conjugation represent a major break-through in modern physics, a great question remains as to whether physical laws are invariant to the third fundamental transformation: time-inversion. Experimental evidence on this point is still not at hand, but it was felt at the conference that it will not be long in coming.

Other important questions-without-answers are not lacking. The principle of conservation of leptons and of microscopic causality, both of which bear on the theory of high-energy processes, are assumptions about which more must be learned. There is the question, stated most succinctly by Pauli, as to why "the Lord is a weak left-hander," referring of course to the fact that violation of parity conservation has been observed so far only in weak interactions. We are provocatively close to some of the answers; perhaps the Eighth Conference (which will be held in the summer of 1958 at Geneva) will bring some of the answers.

* * *

We are indebted to Prof. E. M. Hafner of the University of Rochester for this account of developments revealed at the seventh High-Energy Conference.

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BOOKS

Nuclear Reactor Physics

By RAYMOND L. MURRAY (Prentice-Hall, Inc., Englewood Cliffs, N. J., 1957, xi + 317 pages, $10.00).

Reviewed by A. F. Henry, Atomic Power Div., Westinghouse Electric Corp., Pittsburgh, Pa.

There are two rather extreme viewpoints that can be adopted with regard to the most satisfactory manner of introducing the subject of reactor physics.

Proponents of the first method point out that reactor theory per se is all based on an equation that can be written down. Accordingly, they agree that it should be written down, and that the subject should be presented as a series of deductions from, or more tractable simplifications of, this basic equation. They point out that such a procedure provides a means of judging the validity of approximations that must generally be made if the solution to a practical problem is to be found and that it indicates alternate avenues of approach should the most tractable approximations turn out to be invalid.

Those who subscribe to the second point of view grant all the arguments of the purists but point out that most approximations suggested by transport theory require a significant amount of mathematical analysis and usually must be solved by machine. Under these circumstances the physical concepts needed for a clear understanding of the field are obscured by mathematical detail. Thus, they argue, it is better to start by examining an artificially simple case and to add complexities only after the basic physical ideas are grasped.

The proponents of this second viewpoint will thoroughly enjoy Professor Murray's book. Those who subscribe to the first viewpoint will feel that, at the very least, the order of the chapters should be reversed.

The book succeeds in presenting very clearly and with a minimum of mathematical detail the basic physical ideas of reactor theory. The procedure adopted is to demonstrate these ideas by considering examples for which elementary mathematical models are fairly adequate. Complexities that may require more involved analysis are generally mentioned but are not allowed to obscure the over-all framework of the theory. Thus diffusion theory is presented with reference to weakly absorbing media; slowing-down theory is described in terms of a large weakly absorbing system; the description of cell theory is applicable primarily to a uranium-metal, graphite-moderated core; the reactor kinetics equations developed apply accurately only to a large, uniformly loaded, highly enriched reactor.

The order of presentation is from the particular to the general. Thus Professor Murray first introduces the idea of reaction rate, then proceeds to the description of diffusion theory and age theory. With these tools, criticality is discussed. Next follow accounts of cell theory, two-group theory and time dependent behavior. Problems involving core temperature and reactor control are then discussed. The final two chapters serve to introduce the subject of transport theory and certain approximations to it. Throughout the text very helpful numerical examples are presented, and each chapter is followed by a collection of problems (with answers) and references.

The book provides a clear, concise introduction to the problems of reactor theory and as such should be excellent as a text for an elementary course in the subject. Those who pursue the subject further will find the field has developed so fast that current methods for estimating reactor behavior differ considerably from those described by Prof. Murray. The basic equations and physical concepts that he presents remain, of course, unchanged.

BOOKS RECEIVED

Russian-English Atomic Dictionary, by Eugene A. Carpovich (Technical Dictionary Co., New York, 1957, 317 pages, $12.00). *(To be reviewed.)*

Clinical Use of Radioisotopes, edited by T. Fields and L. Seed (Year Book Publishers, Inc., Chicago, 1957, 455 pages). *(To be reviewed.)*

A Handbook of Hard Metals, by W. Dawihl (Philosophical Library, Inc., New York, 1956, viii + 162 pages, $10.00). This book is a translation and condensation prepared in England from Dawihl's original German manuscript, which was prepared sometime previous to the end of The Second World War. The first part of the book discusses the sintering technique, and the second part, the production of carbides.

Electricity and Magnetism, by J. Newton (Philosophical Library, Inc., New York, 1956, ix + 613 pages, $10.00). The subject is presented on a first-year-college level.

Atoms at Work, by John Mander (George Newnes, Ltd., London, 1957, x + 118 pages, about $1.50). This little book relates, in a manner remarkably innocent of technical jargon, the story of the nuclear field.

Corrosion: A Compilation, by Mars G. Fontana (Hollenback Press, Columbus, Ohio, 1957, 240 pages). This book gathers together monthly columns that have appeared for the past ten years in *Industrial and Engineering Chemistry*. It covers the causes and handling of corrosion problems as encountered with metals, alloys and nonmetals.

Report on the Atom, 2nd ed., by Gordon Dean (Alfred A. Knopf, Inc., New York, 1957, vii + 359 pages, $5.00). A new chapter highlighting nuclear developments during 1953–1956 stands as the essential difference in the second edition of Gordon Dean's book. In 37 pages, Dean, former AEC commissioner, summarizes the various nuclear activities during this three-year period. Besides discussing such matters as thermonuclear energy for industrial purposes, the Geneva conference, President Eisenhower's atoms-for-peace program, economic factors facing domestic power projects and military uses, Dean perhaps makes his strongest point in the area of education: "It is clear that we should be making every possible effort to produce a technological generation to man what is essentially a new technological age. We are, I am afraid, failing very badly."

ALSO OF NOTE

Proceedings of the First Annual Conference of the Health Physics Society. Copies are available. *C. C. Palmiter, Editor, Health Physics Society, Radiation Control Service, University of Michigan, Ann Arbor, Mich., $3.00.*

Radiological Health Handbook. This Public Health Service handbook, edited by Simon Kinsman, was originally compiled for training courses at the Robert A. Taft Sanitary Engineering Center. Contents include a glossary and tables, charts, graphs and nomograms in sections giving physical,

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(In the Suburban San Fernando Valley, Los Angeles)

chemical and mathematical data; decay and radioassay data; radiation-protection data; and isotope properties. *Office of Technical Services, U. S. Dept. of Commerce, Washington, D. C., $3.75.*

Bibliography on Industrial Radiology. The sixth supplement is available, covering the years 1954–56. The second edition of the original work was published in 1943. *St. John X-Ray Laboratory, Califon, N. J.; 6th supplement, $3.00; all 6 supplements in one volume, $10.00.*

What's Available in the Unclassified Atomic Energy Literature (TID-4550). The third revision of this guide tells what information is available and how to get it. *Technical Information Service Extension, U. S. Atomic Energy Commission, P. O. Box 62, Oak Ridge, Tenn., no charge.*

Russian-journal translations. The physical chemistry and applied physics sections of the Proceedings of the Academy of Sciences of the U. S. S. R. are available in English translation. Annual subscriptions are $160 and $200, respectively. *Consultants Bureau, Inc., 227 W. 17th St., New York 11, N. Y.*

Scientific German. This handy little book, by George E. Condoyannis, is intended, when augmented by a dictionary, to make technical German understandable to the reader with no previous knowledge of the language. It does not promise to engender proficiency in speaking or writing. 164 p. *John Wiley and Sons, Inc., 440 Fourth Ave., New York 16, N. Y., $2.50.*

Scientific French. William N. Locke has written this book to fulfill the same purpose as "Scientific German" (see above). 112 p. *John Wiley and Sons, $2.25.*

Effect of Irradiation on Solids. This bibliography, compiled by Helen C. Friedemann, is an expanded fourth edition of the bibliography originally compiled by Penn-Texas Corp. and issued by the AEC. References are grouped as theory, metals, and nonmetals. *Penn-Texas Corp., Nuclear Engineering Div., New York 19, N. Y., and The Martin Co., Nuclear Div., Baltimore 3, Md.*

Proceedings of the Instrument Society of America's National Nuclear Instrumentation Conference. The proceedings of the April '57 meeting include sessions on nuclear instrumentation for

reactors, industry, health physics, university research and the medical field and a session on basic instrumentation problems. *Instrument Society of America, 313 6th Ave., Pittsburgh 22, Pa., $5.00.*

1957 Nuclear Congress Papers. Individual papers of the Nuclear Engineering and Science Conference will be available until Jan. 15, 1958, at 30¢ each. *American Society of Mechanical Engineers, 29 W. 39th St., New York 18, N. Y.*

LETTERS

Addenda

DEAR SIR:

I regret to call to your attention an error in the data table accompanying the SPERT-I stability article (NU, Jan. '57, p. 41). Transient 3068 should read 1.75% reactivity addition rather than 1.5% as printed.

—S. G. FORBES, Supervisor
Physics Section
Reactor Projects Branch
Atomic Energy Division
Phillips Petroleum Co.
Idaho Falls, Idaho

DEAR SIR:

I would like to call your attention to the misuse of our trademark "Thiokol" (NU, July '56, p. 28, Oct. '56, p. 51). The word Thiokol is a registered trademark for products manufactured by Thiokol Chemical Corp. and includes not only polysulfide rubber, but also polyesters, plasticizers, rocket motors and propellants. It is because of possible harm to our trademark that we feel it essential that we request future reference to our materials be more accurately made, such as polysulfide rubbers.

—F. O. DAVIS, Supervisor
Patent Section
Thiokol Chemical Corp.
Trenton, N. J.

Reprints Available

Computers in Nuclear Engineering (May 1957)—$1.25

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NUCLEONICS

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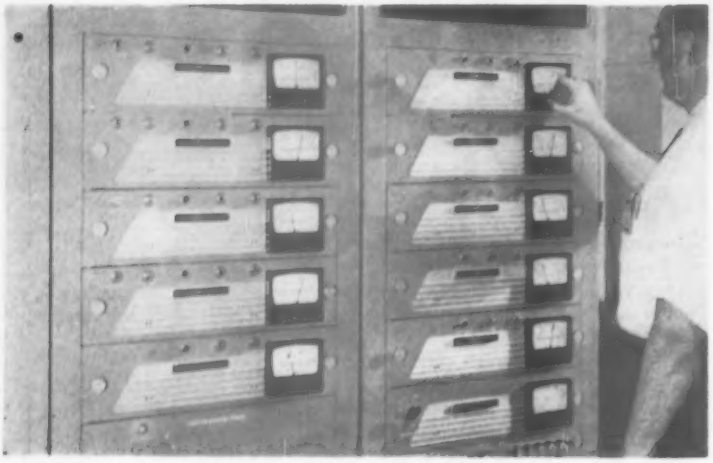
PRODUCTS AND MATERIALS



Adjustable Meter-Relays to Monitor Shippingport Reactor

Meter-relays shown in the monitor panel for the Shippingport Pressurized Water Reactor will flash warning signals, actuate audible alarms, and operate valves if radiation levels exceed tolerance values. Meters combine constant visual indication with easily adjusted preset relay contacts. Each has a 3-decade logarithmic scale.

Monitors will (1) check for leakage in water cooling system, (2) check radiation levels in boiler compartments to make sure they are safe for maintenance, (3) check ventilating and cooling air for contamination, and (4) keep chart records of radiation levels. If air systems are contaminated, meter-relays will actuate solenoid-operated valves to control circulation.

Monitoring system is housed in auxiliary control room. Electronic circuitry for 11 channels is mounted in roll-out slides behind removable doors. All channels but one have low-level flashers to warn of readings that might indicate a malfunction of a monitor. Special features include etched wiring, plug-in circuits, circuit uniformity, tubeless power supplies, spare low-voltage and filament power supplies.—Assembly Products, Inc., Chesterland, Ohio.

Neutron Analyzer

Neutron time-of-flight analyzer has 1,024 channels with widths adjustable from minimum of $\frac{1}{2}$-μsec to higher values. Records 65,535 counts/channel with dead time of 16 μsec/count. Memory can be divided into two 512-channel analyzers for recording simultaneous but separate events. Magnetic-core memory has 92 vacuum tubes; control circuit has 80 tubes. Power consumption is 20 amps at 115 volts.—Radiation Counter Laboratories, Nucleonic Park, Skokie, Ill.

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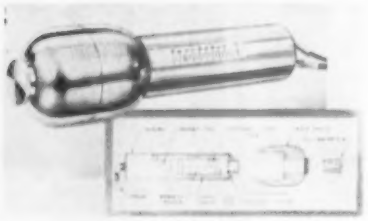
Borescope. Can see inside reactors from a safe distance. Device can be inserted through openings as small as $\frac{1}{16}$ in. Unit has long tube with tiny incandescent lamp, objective lens in one end, and eyepiece with ocular lens and erecting prism in other.—National Electric Instrument Co., Elmhurst, N. Y.

RADIATION DETECTORS AND ACCESSORIES



Scintillation Detector

Designed for operation with scalers, ratemeters or gamma ray spectrometer, SD-1 has choice of three collimators gives selection of focusing, depth and field scanning.—Technical Associates, 140 W. Providencia Ave., Burbank, Calif.

TV Camera

Miniature TV camera can examine remote locations. Measuring 5¼ in. long and 2 in. dia, this camera has built-in light source. Unit can be maneuvered with perlon cable which houses control wires.—Majestic International Sales Corp., 79 Washington St., Brooklyn, N. Y.

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NUCLEAR NEWS FROM ATOMICS INTERNATIONAL

AI develops reactors for largest power plants and smallest research labs

A wide selection of nuclear reactors, ranging in size from a 75-megawatt nuclear power plant to generate electricity, to a 5-watt laboratory instrument, have already been completed, or are at present under construction by ATOMICS INTERNATIONAL.

The Sodium Reactor Experiment, part of the Atomic Energy Commission's program to develop economical nuclear power, now undergoing tests in the Santa Susana Mountains near Los Angeles, will have an output of 20 thermal mw and will generate about 6.5 mw of electric power. It is the prototype of a full-scale sodium graphite reactor design, which will have a power output of 75 to 125 mw. ATOMICS INTERNATIONAL regards it as a promising reactor type for economical power generation.

The Organic Moderated Reactor, a relatively new approach to economical nuclear power, is particularly appropriate for smaller plants. ATOMICS INTERNATIONAL is completing construction of the Organic Moderated Reactor Experiment for the Atomic Energy Commission at the National Reactor Testing Station in Idaho, part of a development program for the OMR. Basic advantages of the OMR include compact core with good neutron economy, low pressure coolant system, and freedom from corrosion problems. The OMR is also considered promising for ship propulsion, and ATOMICS INTERNATIONAL is making a design study of this reactor type for maritime use.

Research Reactors are now being built by ATOMICS INTERNATIONAL for Japan, Denmark, West Germany, and West Berlin. They are similar to the ATOMICS INTERNATIONAL reactor now in service at Armour Research Foundation, Chicago—the first to be designed specifically for private industrial research.

The Laboratory Reactor is a new, low-cost instrument designed specifically to meet the needs of universities and industrial laboratories for nuclear training and research.

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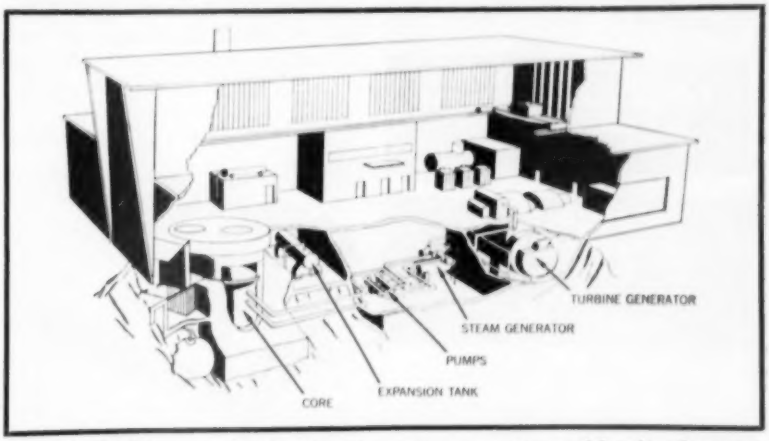
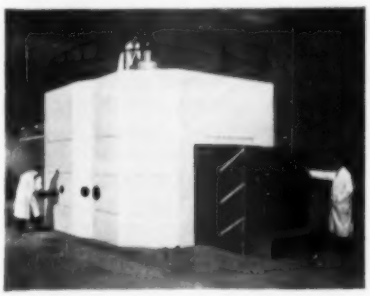
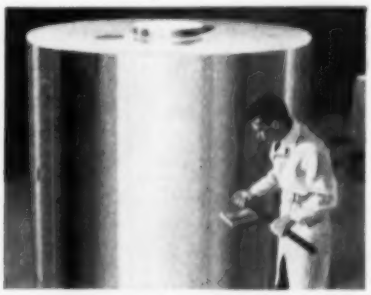
Reactor "heart". An AI technician installs zirconium "cans" containing the graphite moderator in the core of the Sodium Reactor Experiment. The use of a sodium coolant allows a high-temperature, low-pressure heat extraction system with high thermal efficiency of power conversion.



Typical OMR Central Station Power Plant. Basic advantages of OMR include small size, compact core, low-pressure system, and freedom from corrosion problems. This also makes it a promising reactor for ship propulsion.



Armour Reactor is a 50 kw water-boiler unit for industrial research. It has nine beam tubes for experimental purposes.



Nuclear Nutshell. Laboratory Reactor operates at 5 watts of power and produces more than 100 isotopes. It can be operated by one trained man.

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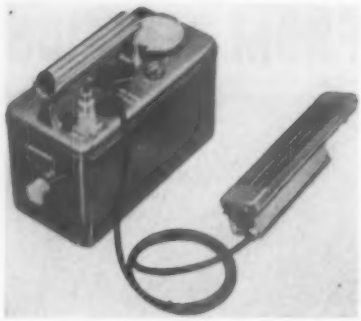

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Model 2112-P, portable survey instrument for alpha surface contamination measurement, consists of unsealed air proportional alpha probe connected by cable to battery operated count rate meter. Meter furnishes three ranges: 0–150, 0–1,500, and 0–15,000 cpm. Unit will measure α contamination from 2 cm²/min to 2,000 cm²/min. Probe efficiency is ~25% for 5-Mev alphas from Rad + E and from U_3O_8 for 2π geometry.—Nuclear Chicago, 223 W. Erie St., Chicago 10, Ill.

Low-Power Counters

Model 100L operates from plate supply of 150 volts d-c at 7.5 ma from 0–100 kc. Specifications for low-power decade counting units include: 100,000-pulses/sec counting rate; 5-μsec pulse-pair resolution. Unit has 35–75 volts negative pulse input requirement with rise time of 0.8 μsec and duration of 2 μsec minimum.—Computer-Measurements Corp., 5528 Vineland Ave., North Hollywood, Calif.

Pulse-Height Analyzer

Quartz line pulse-height analyzer with 100 channels makes use of pulse-height-to-time converter unit. It records spectrum data on long and short lived radioactive isotopes. Analyzer consists of expander amplifier, pulse stretcher, delay-line storage element and means for converting pulse height to time measurement.—Nucleonic Corp. of America, 196 DeGraw St., Brooklyn 31, N. Y.

Gamma Switch

Unit is switch rather than precision measuring instrument. Consists of halogen-quenched G-M tube in circuit with simple relay. With dimensions of 11 × 10 × 4-in, switch operates only when radiation received increases-decreases beyond predetermined level.—Isotope Developments Ltd., Beenham Grange, Aldermaston Wharf, England.

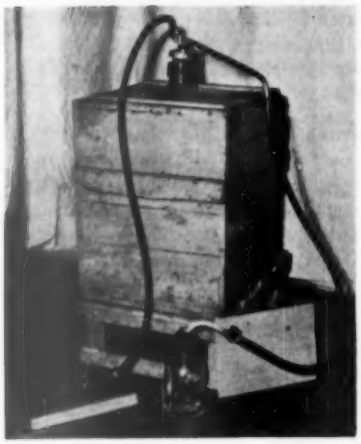
•

Prober. Model B60, a subsurface probe, is based on Geiger principles. Has sensitivity of ~500,000 cpm/mr/hr when exposed to radium γ; can detect as little as 0.0005 mr/hr. Unit not affected by small voltage changes and operates from −40° to more than 212° F. Length is 13 in; diam. 2¾ in.; $375.—Nucleonic Corp. of America, 196 DeGraw St., Brooklyn 31, N. Y.

Gas chromatograph. Model GC-2 analyzes gases and liquids with boiling points up to 350° C with 0.1% full scale accuracy.—Beckman Instruments, Inc., Fullerton, Calif.

Remote area monitoring. Unit is made for monitoring radiation from single control center to as many as 20 remote stations. System is comprised of basic control unit, plug-in stations, remote sensing unit with sealed ionization chamber, calibration check source with strontium-90 beta emitter, and beta window with discriminator. Sensing elements can be up to 5,000 ft from control point; system has ±15% of radiation rate at any point within detector's range.—Victoreen Instrument Co., 5806 Hough Ave., Cleveland 3, Ohio.

LABORATORY INSTRUMENTS AND APPARATUS



Pump Dispenser

By elevating carboy above receiving container and creating flow with air pump, syphon is formed. Tube length is 56-in.—General Scientific Equipment Co., 7516 Limekiln Pike, Philadelphia 50, Pa.

Fluorimeter

Instrument quantitatively analyzes uranium minerals by means of optical fluorescence. Light source excites sample, and photomultiplier detects fluorescence. Reads directly amounts as small as 10^{-10} gm.—Mt. Sopris Instrument Corp., 1320 Pearl St., Boulder, Col.

●

High-temperature source. Plastmatron produces and maintains controllable temperature up to 26,540° F. Unit is available in two sizes—80 kw, 440 volts; 160 kw, 440 volts or up to 10,000 kw. Heat source is plasma jet, adjustable gas jet using various noble gases that is ionized by electric charge. Has visible flame from 1–6 in.—Plasmatron, 18,400 S. Main St., Santa Ana, Calif.

Transistor furnace. Electric furnace is designed for alloying, brazing, soldering transistors, other semiconductors. —BTU Engineering Co., 440 Somerville Ave., Somerville 43, Mass.

Ultrasonic generators. Glennite U-405 unit drives low-impedance ultrasonic transducer; has 500-watt r-f power output. It operates at 40-kc fixed frequency or at any selected frequency between 20 kc and 2 Mc.—Gulton Industries, Inc., 212 Durham Ave., Metuchen, N. J.

NEW MATERIALS

Chromates Prevent Corrosion of Aluminum in Pool Reactor

Chromate-conversion coating on underwater aluminum parts prevents corrosion during submersion in a pool-type reactor. Coating is not noticeably affected by neutron bombardment. The finish—Iridite #14-2—has been applied to underwater parts in the U. S. Naval Research Laboratory's reactor. Coating is used on 0.02-in. aluminum fuel cladding as well as on core structure. Doors leading into pool below water level and beam portholes in shield assembly are also coated. Aluminum components are in direct contact with other metals at some points, yet coating minimizes galvanic corrosion.—Allied Research Products, Inc., Baltimore, Md.

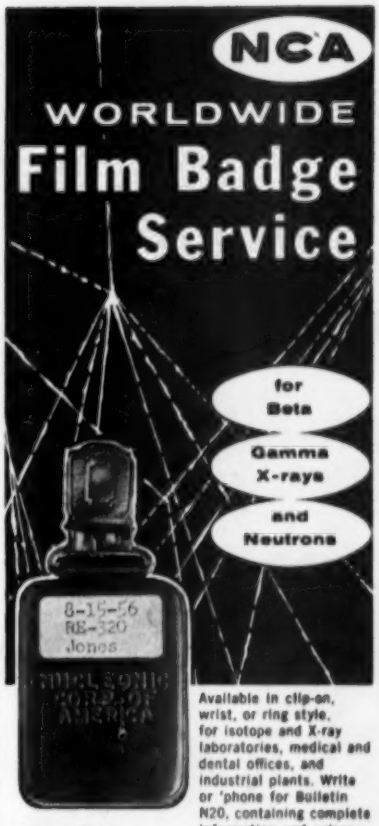
Boron Steel

Austenitic alloys 1% boron stainless has improved thermal-neutron absorption properties. Available in plate, sheet, strip and other forms, it has these mechanical properties when thickness ranges from 0.005–1 in: tensile strength, 90,000 psi; 0.2% yield strength, 50,000 psi. Boron stainless is essentially two-phase alloy composed of complex boride phase in austenitic chrome-nickel-iron matrix. Material has been tested in water at 550° F for extended periods for suitability in EBWR.—Superior Steel Corp., Carnegie, Pa.

●

Refractory cement. Metal, glass, or ceramics will be held to metal despite temperatures between −420° and 1,000° F. Called CA-9, the adhesive can be extended to 1,500° F by modifying composition. Material is waterproof, nonhydroscopic and unaffected by most reagents. Cement composed of ground phlogopite, polysiloxane and refractory oxides.—Charles Engelhard, Inc., 850 Passaic Ave., East Newark, N. J.

C^{14} compounds. Sixteen C^{14}-labeled organics, most with specific activities of ~1.0 mc/millimole have been added to manufacturer's list.—Tracerlab, 130 High St., Boston 10, Mass.



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REACTOR COMPONENTS

Centrifugal Blower

Leak-proof blower handles corrosive, toxic and radioactive gases. Designed to operate at gas temperatures of 500° F, unit will accommodate 20 cfm of gas having specific gravity of 6 and will develop 150-ft heat in 2,000-psi system. Consists of centrifugal blower and canned motor sealed in heavy stainless steel enclosure. Radioactive contamination and water corrosiveness in rotor and bearing cavity are limited by labyrinth seals at water-gas interface. —Allis-Chalmers Mfg. Co., Box 512, Milwaukee 1, Wis.

•

Sodium-level transducers. Units are designed to sense level height of liquid sodium and other liquid metals. Level ranges are available from few inches to 30 ft.—Crescent Engineering & Research Co., 5440 N. Peck Rd., El Monte, California.

CONTROLLERS AND RECORDERS

Strip-Chart Recorder

Model G-11, self-balancing strip-chart recorder tabulates to ±1%. Rectilinear strip chart is 5-in. wide in 85 ft rolls. Speeds from ⅛ in./hr to 8 in./min. Weight 15 lb; cost $395.—Varian Associates Instrument Division, 611 Hansen Way, Palo Alto, Calif.

•

Chassis for controls. Made for Simplytrol and Versatrol package controls, new chassis style saves space in built-in flush-mounted applications. Chassis are 4¼ in. wide, 7⅜ in. deep.—Assembly Products, Inc., Chesterland, Ohio.

Telemetering. Called "Electro-Span," system has instruments to measure pressure, temperature, fluid level; relays data digitally.—Bendix Aviation Corp., 11,600 Sherman Way, North Hollywood, Calif.

Pressure transducer. Has pressure ranges from 0–2 to 0–250 psi. Offered with potentiometer cards in resistance ranges from 0–2,000 to 0–20,000 ohms. Unit is 2 1/16 in. long, 2⅛ in. dia., weighs 6 oz.—Bristol Co., Waterbury 20, Conn.

LITERATURE AVAILABLE

Capacitors. Specifications of Electromite capacitors are described in engineering bulletin 533.—Cornell-Dublier Electric Corp., South Plainfield, N. J.

Potentiometer instruments. Four page bulletin discusses electronic Speedomax G potentiometer instruments for indicating thermocouple temperatures.—Leeds & Northrup Co., 4934 Stenton Ave., Philadelphia 44, Pa.

Plugs. Polyethylene caps and plugs are discussed in folder.—Protective Closures Co., 2207 Elmwood Ave., Buffalo 23, N. Y.

Facilities. Manufacturing of electronic test equipment components is outlined in 12-page brochure.—Laboratory for Electronics, Inc., 75 Pitts St., Boston 14, Mass.

Rectifiers. Physical and electrical characteristics of 10 silicon power rectifiers are given in data sheet 40.—Hughes Products, International Airport Station, Los Angeles 45, Calif.

Potentiometers. Data sheet 54-23 discusses linearity tolerances in precision potentiometers; AN and CN precision potentiometers are described in data sheet 54-12; technical paper 762 reports on the "Environmental Effects on Precision Potentiometers."—Helipot Corp., Newport Beach, Calif.

Densimeter. Series 27 densimeter is subject of 4-page report.—Potter Aeronautical Corp., Union, N. J.

Radioactivity safeguards. Health and safety precautions against hazards of radioactivity can be found in bulletin 407.—Nuclear Science & Engineering Corp., PO Box 10910, Pittsburgh 36, Pa.

Recordings. Six page folder shows oscillographs, amplifiers, Penmotor, other direct-writing recording systems.—Brush Electronics, 3405 Perkins Ave., Cleveland 14, Ohio.

Laboratory glass. Laboratory burettes are listed in catalog 80C100—Fischer & Porter Co., 10 Jacksonville Rd., Hatboro, Pa.

Analytical instruments. Laboratory analytical instruments—spectrophotometers, monochrometers and fractrometers—are noted in 16-page

booklet.—Perkin-Elmer Corp., Norwalk, Conn.

Career opportunities. "Your Opportunities in Industry as a Technician," is title of 32-page booklet on ways to select and train for technical fields.—National Association of Manufacturers, 2 East 48 St., New York 17, N. Y.

Hydraulic valve. Catalog sheet 811 describes spring-loaded, diaphragm-actuated hydraulic pilot valve.—Atlas Valve Co., Newark 5, N. J.

Metal-forming machines. Both hot- and cold-forming equipment, are discussed in 4-page pamphlet.—Fenn Nuclear Machinery Division, Fenn Rd., Newington, Conn.

Computer. Pamphlet describes use of electronic computation and data processing.—Data Control Ltd., 880 Bay St., Toronto 5, Ont.

Digital indicator. Two-page data sheet discusses computing digital indicator—Dynac, Inc., 395 Page Mill Rd., Palo Alto, Calif.

Uranium demand. Atomic letter 25 reports on uranium requirements in 1967.—Atomic Development Fund, Inc., 1033–30th St., N. W., Washington 7, D. C.

Heavy construction. "Engineering and Construction for Industry and Government" is the title of a 48-page booklet.—Ralph M. Parsons Co., 617 S. Olive St., Los Angeles 14, Calif.

Tungsten carbide. Characteristics of tungsten carbide with platinum binder for corrosion conditions are presented in 2-page bulletin.—Kennmetal Inc., Latrobe, Pa.

Radiation protection. Four-page booklet describes lead shielding material.—Ameray, Rte. 46, Kenvil, N. J.

INDUSTRY NOTES

▶**Carborundum Metals Co.** will begin production of reactor-grade zirconium in its new plant sometime this summer.

▶**Nuclear Magnetics Corp.**, a **Perkin-Elmer** subsidiary, has moved from Boston to parent company headquarters in Norwalk, Conn.

▶**Isotopes, Inc.**, has formed **Crystals, Inc.**, a new subsidiary, to do research on all phases of crystal growing. Facilities will be in Westwood, N. J.

▶**Servomechanisms'** Magnetics division has completed its new plant in Hawthorne, Calif.

▶**National Research Corp.** has promoted James H. Gardner and Arthur G. Walsh as assistant directors in its research division.

▶B. F. Armbruster joined **Nuclear-Chicago Corp.** as treasurer and controller.

▶F. Lee Foster, Jr., previously with High Voltage Engineering Corp., has joined **Baird-Atomic** as vice-president and sales manager. He was succeeded by E. Alfred Burrill, who has been with High Voltage since 1947.

▶John Carter, Corning Glass Works vice-president, elected president, **Fairchild Camera & Instrument Corp.**

▶**Applied Radiation Corp.** has opened its plant in Walnut Creek, Calif., to manufacture electron linear accelerators.

▶**Norton Co.** promoted William F. Winemiller to manager of sales engineering department in its refractories division.

▶**National Distillers & Chemical Corp.** is the new name of **National Distillers Products Corp.** A subsidiary, **U. S. Industrial Chemicals Co.**, will retain its present name.

▶**Assembly Products, Inc.**, has elected John D. Saint-Amour as president and Robert H. Pugsley as sales vice-president. They succeed Bradley R. Thompson, who held both posts; he was elevated to board chairman.

▶Duane M. Mayhew **Victoreen Instrument** general sales manager, was elevated to sales vice-president. William A. McCarthy, recently with Tracerlab, was named sales manager.

▶**International Business Machines Corp.** has opened its first radioisotope laboratory in Endicott, N. Y. to study and analyze components of its data processing machines.

▶**Union Carbide Corp.** is the new name of Union Carbide & Carbon Corp.

▶**Nuclear Corp. of America** was acquired by **Baker & Co.**, subsidiary of **Engelhard Industries.** Nucor will remain under present management.

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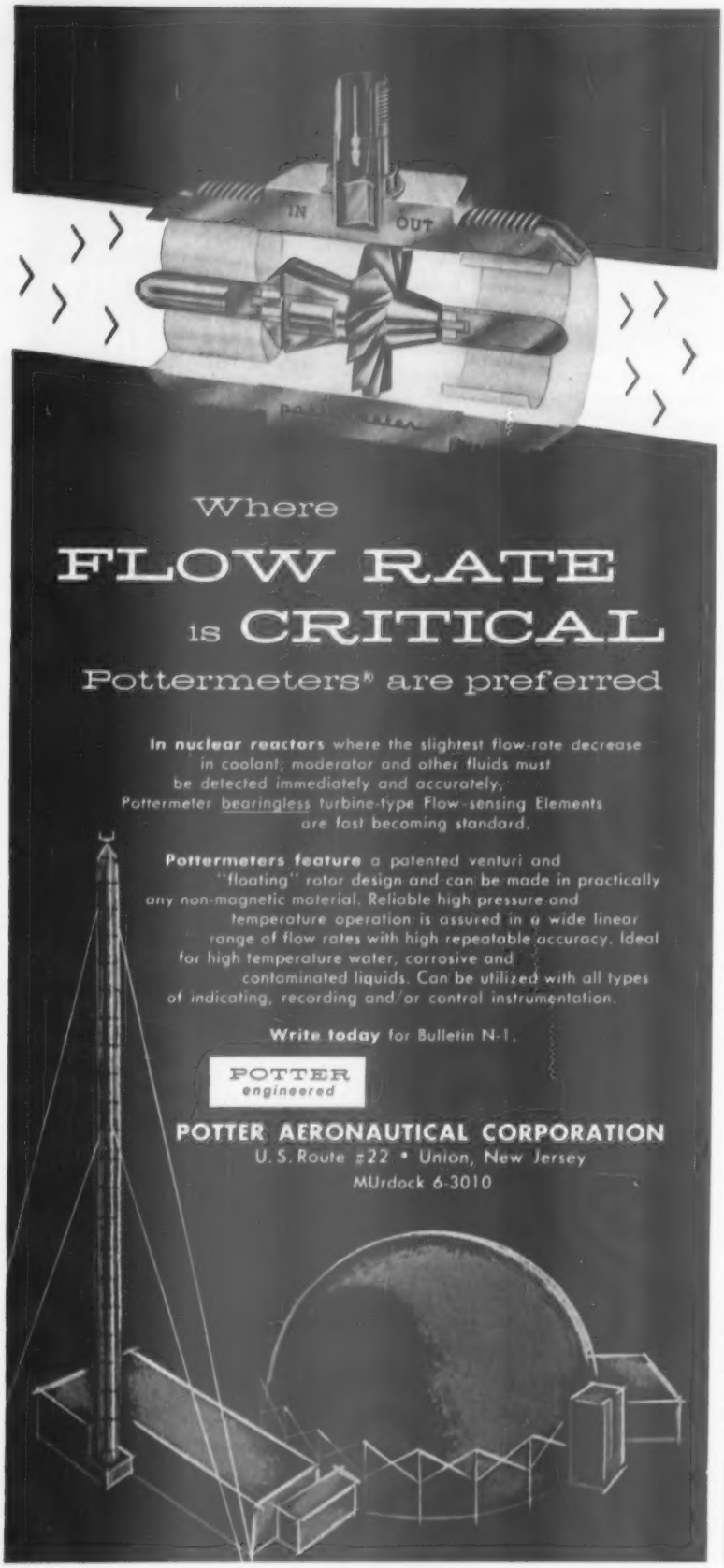
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NEWSMAKERS

Robert W. Kirkman, with AEC's New York operations for the past 10 years as security head, has been named inspection director for that office.

Clevite Research Center's nuclear metallurgy section has appointed **Robert D. Johnson** as its head. He was previously with Battelle Institute.

Henri Polak is Atomics International's new European representative.

Murray Medvin, former AEC assistant general counsel, has joined Gulton Industries as executive assistant to the president.

Argonne National Laboratory has appointed **Carl W. Larsen** executive assistant for public information and **Lynn K. Hurst** director of the special materials and services division.

John C. Herrera and **Edwin Westbrook** have joined High Voltage Engineering Corp. as senior engineers. Herrera was previously chief engineer with New London Instrument Co., and Westbrook was with Federal Telecommunication Laboratories.

Shell Development Co. has named **D. P. Stevenson** as associate director of its fundamental research group. He will be succeeded as head of the chemical physics department by **J. W. Otvos.**

National Distillers & Chemical Corp. has elected **Robert E. Hulse,** general manager of its chemical division, to be executive vice-president—a new post. Also **Robert H. Cornwell, Alden R. Ludlow, Jr., Francis Olmstead** and **Stuart Schott** were named chemical division vice-presidents.

Union Carbide Co.'s research center in Sterling Forest, N. Y., will be administered by **H. Rush Spedden,** research director, Union Carbide Ore; and **F. W. Hurd,** research director, Union Nuclear.

U. S. Radium Corp. promoted **C. C. Carroll** to be research coordinator. He is succeeded by **H. A. Vaughan** as manager of the Bloomsburg, Pa., plant.

Alexander Hollaender, director of the biology division at Oak Ridge National Laboratory, was elected a member of the National Academy of Sciences.

John W. Bond, Jr., was appointed head of Convair's (San Diego) physics group, a new section to study nuclear propulsion.

Radiation Counter Laboratories of Skokie, Ill., promoted **Robert W. Schumann,** chief development engineer, to vice-president for research. **Vernon L. Brown,** formerly chief engineer, succeeds Schumann.

H. G. Raggatt was appointed deputy chairman of the Australian Atomic Energy Commission. He is secretary of the Department of National Development, and has advised **Australia's** AEC on uranium mining since 1953.

Daniel I. Weinberg, Gerald H. Katzin and **Earl M. Page** have joined Advanced Scientific Techniques Research Associates (ASTRA) of Milford, Conn.

Battelle Memorial Institute has promoted **Edward E. Slowter** to vice-president.

Armour Research Foundation has appointed **Frank Genevese** as assistant manager in its propulsion and fluid mechanics research department. He had done nuclear physics work at Kellex Corp.

Tracerlab has elected **Samuel S. Auchincloss** as president and board chairman, succeeding **William O. Faxon** who resigned. Auchincloss had been head of American Machine & Foundry's Cleveland welding division. Faxon had succeeded to the top spot only a year ago.

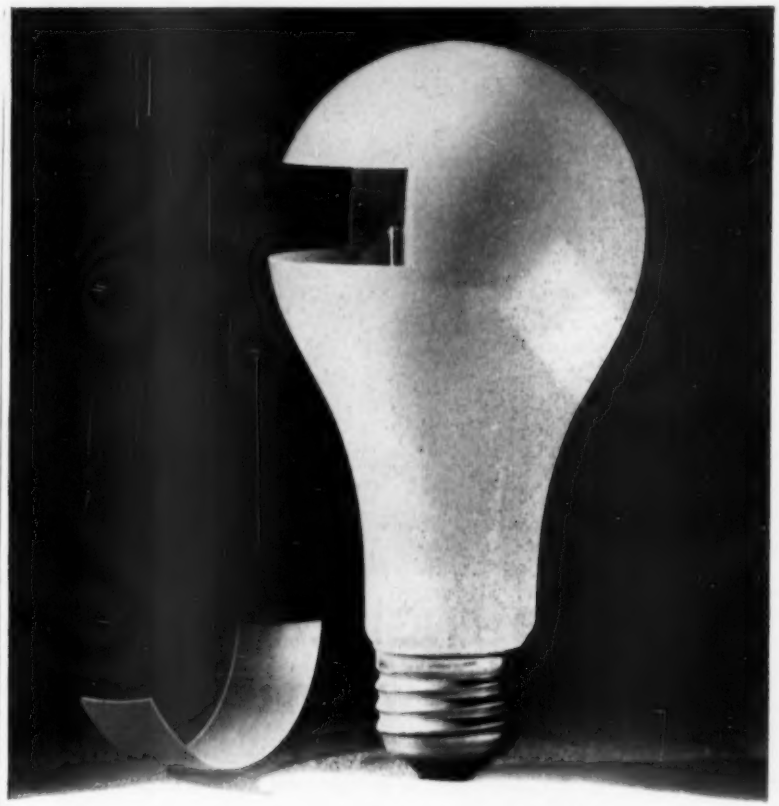
Paul C. Aebersold, director of AEC's Isotopes Extension at Oak Ridge from its establishment in 1947 until its transfer to Washington last year, has been named Assistant Director for Isotopes and Radiation of the Division of Civilian Application, with offices in Washington.

Stewart M. Lang, ceramic engineer at the National Bureau of Standards, was awarded a Department of Commerce silver medal for his work on the high temperature physical, chemical, and thermodynamic properties of ceramic materials in their nuclear applications.

Edward A. Mason has left Ionics, Inc. as research director to become professor of nuclear engineering at Massachusetts Institute of Technology.

National Research Council of Canada has named **Harry Williamson** as its scientific liaison officer in Washington, D. C.

Clyde L. Cowan, Jr. was appointed professor of physics, George Washington University, after eight years' service with AEC in Los Alamos.



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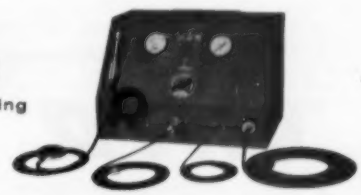

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Vitro Engineering, now well engaged in foreign activity, is also at work in Europe, where it is designing Italy's principal nuclear research center.

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NUCLEAR CAMPUS

• Columbia University has set up a **Council for Atomic Age Studies,** objective of which is to make Columbia a center for the study of problems facing society as a result of the development of atomic energy. As these problems overlap many fields of activity and of learning, so the Council's work will cut across many fields including physics, engineering, medicine, international relations, journalism, business, philosophy, and law. It will "stimulate, organize and supervise" studies. Possible projects include issues in international development of nuclear power that involve research in technology, law, economics and foreign policy; and joint studies on the effect of new forms and sources of power on industry and agriculture, on physical and mental health, on our domestic economy and our communities.

Courses

A **radiation-biology** summer course will be held July 1–26 at Argonne National Laboratory for persons with biological research experience. Course fee is $25.

Nuclear reactors and radiations in industry will be offered this summer Aug. 19–30 at the Univ. of Michigan's College of Engineering, Ann Arbor.

Massachusetts Institute of Technology will offer **radioisotope utilization principles** July 8–19.

Radiation for industrial physicians and lawyers is subject of a one-week course starting Sept. 9 at U. of Cincinnati's Institute of Industrial Health and College of Law. The course is directed at the two professions simultaneously to enable each to understand the problems, professional responsibilities and limitations of the other when dealing with medical and legal problems of radiation. Tuition is $100. Write Secretary, Institute of Industrial Health, U. of Cincinnati, Cincinnati 19, O.

Discussion **seminars on radioisotopes** for industrial management and technical personnel are being offered by Isotopes, Inc., at its labs, 123 Woodland Ave., Westwood, N. J. They will be scheduled at times suitable for interested companies, at no cost or obligation. Aim is to help management become familiar with potential isotope uses in their sphere of interest, and to learn of new developments in the field. Write D. R. Carr, Isotopes, Inc.

NUCLEAR CALENDAR

June 10–12—Third annual meeting, American Nuclear Society, Pittsburgh (*Wm. Penn-Sheraton*). Contact W. W. Grigorieff, exec. secy., Box 963, Oak Ridge, Tenn.

June 12–14—Second annual radiological safety conference, co-sponsored by Canisius College and N. Y. State Labor Dept., at Canisius College, Buffalo. Designed to provide up-to-date information and technical assistance needed by users of radiant energy in industry to insure maximum protection of workers; exhibits will relate mechanical aspects of industrial use of radiation to health and safety hazards. Contact State Industrial Commissioner Isador Lubin, 80 Centre St., NYC 13.

June 15–Sept. 8—International exhibition "The Atom," near Schiphol airport, Amsterdam. Ten countries including US and USSR will participate.

June 16–21—Annual meeting, Am. Soc. for Testing Materials, Atlantic City (*Chalfonte-Haddon*). Contact Fred F. Van Atta, 1916 Race St., Phila. 3, Pa.

June 17–19—Third annual meeting, Health Physics Society, Pittsburgh (*Pittsburgh U.*). Contact L. C. Emerson, ORNL, Oak Ridge, Tenn.

June 20–22—Fourth annual meeting, Society of Nuclear Medicine, Oklahoma City (*Skirvin*). Contact R. W. Lackey, 452 Metropolitan Bldg., Denver, Colo.

June 27–28—Meeting on powder metallurgy, Düsseldorf, Germany. Contact Board for Powder Metallurgy, Prinz-Georgstrasse 77, Düsseldorf.

July 1–6—First international colloquium on corpuscular photography, under auspices French National Center of Scientific Research, Strasbourg, France. Contact P. Cuer, director, Corpuscular Physics Laboratory, University of Strasbourg.

Aug. 7–9—Sixth annual conference on industrial applications of X-ray analysis, Denver Research Institute (*Albany*). Latest developments in X-ray absorption, diffraction and fluorescence analysis. Contact James P. Blackledge, DRI, Univ. of Denver.

Aug. 20–22—Conference on liquid scintillation counting, Northwestern Univ., at Technological Institute, Evanston, Ill. Coincidence counters, single-channel counters, chemistry of liquid scintillators, applications in biology, medicine, industry, archaeology, physics, chemistry and engineering. Contact F. Newton Hayes, Los Alamos, N. M.

Sept. 9–13—International conference on nuclear structure, co-sponsored by Unesco and International Union of Pure and Applied Physics, at Rehovoth, Israel. Contact Amos de Shalit, nuclear physics department, Weizmann Institute of Science, Rehovoth.

Sept. 9–14—Colloquium on nuclear interaction of neutrons, International Union of Pure and Applied Physics, at New York. Contact W. W. Havens, Pupin Laboratory, Columbia Univ., NYC 27.

Sept. 9–20—International conference on radioisotopes in scientific research, Unesco, in Paris. Purpose: exposition and discussion of new ideas or methods for utilization of radioisotopes in scientific research. Plenary sessions on production and measurement techniques, section meetings in physical and biological applications. Contact Dept. of Natural Sciences, Unesco, 19 Avenue Kléber, Paris 8e.

Sept. 22–27—Colloquium on mesons and recently-discovered particles, International Union of Pure and Applied Physics, in Venice. Contact Prof. Eligio Perucca, director, Politecnico, Turin, Italy.

Oct. 6–12—Ninth atomic energy course for management, National Industrial Conference Board, in Chicago (*Hotel Moraine-on-the-Lake*), Highland Park, Ill. Contact John McMahon, NICB, 460 Park Ave., NYC.

Oct. 28–Nov. 1—Autumn Nuclear Week in New York. Included are: **28–30th,** fourth annual meeting of the Atomic Industrial Forum (meetings *Plaza* and *Coliseum;* banquet *Waldorf*). **28–31st,** third annual Trade Fair of the Atomic Industry, sponsored by AIF (*Coliseum*). **28–30th** second winter meeting of the American Nuclear Society (*Henry Hudson*). **31st,** AEC-ANS-AIF reactor safety conference. And IRE's Professional Group on Nuclear Science holds fourth annual meeting **31–1st** (*Henry Hudson*).

Nov. 2–8—Second world metallurgical congress, American Society for Metals, Chicago. Contact W. H. Eisenman, congress secretary general, ASM, Cleveland. **3–8th,** second international conference on nondestructive testing. Contact G. H. Tenney, 704 47th St., Los Alamos, N. M.

Nov. 7–8—Seventh annual symposium, Philadelphia section of Instrument Society of America, Philadelphia (*Sheraton*). Subject: new ideas in instrumentation evolved from nuclear developments. Instruments Fair included. Contact Warren Brand, Conoflow Corp., 2100 Arch St., Philadelphia 3.

Dec. 24–26—Conference on nuclear sizes and density distribution, National Science Foundation, Palo Alto, Calif. Contact Robert Hofstadter, Stanford Univ., Palo Alto.

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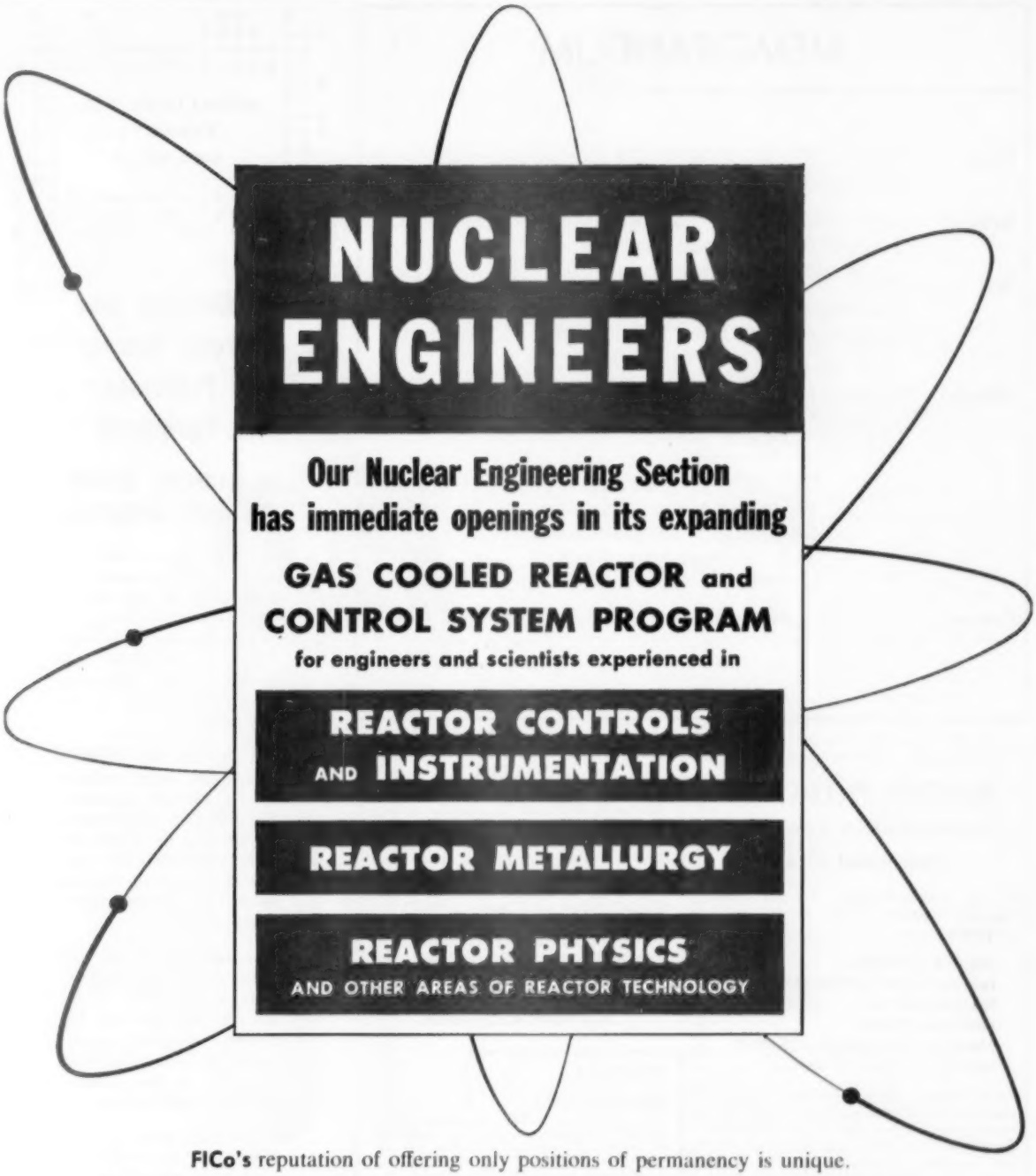

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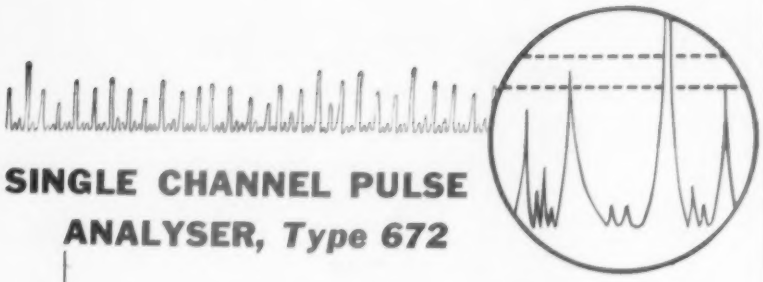

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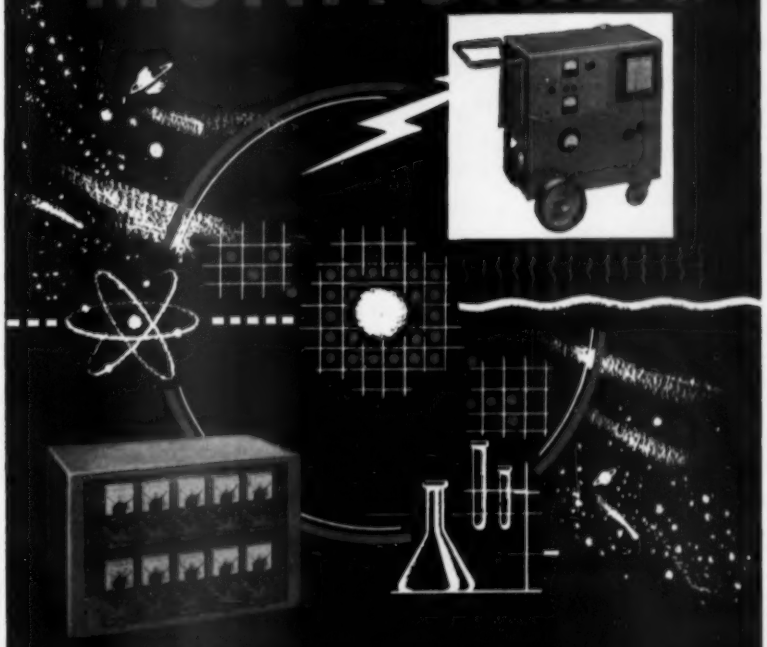

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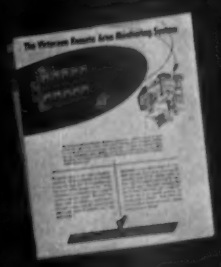

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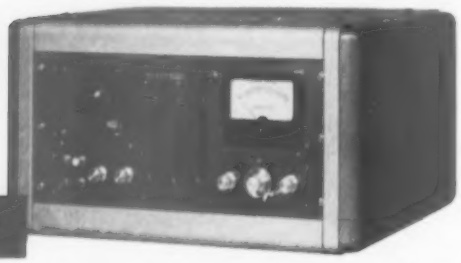

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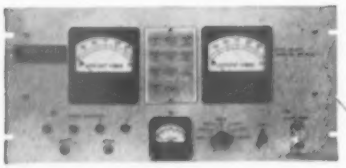

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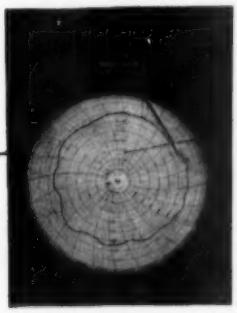

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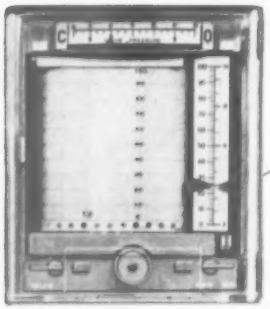

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